LILCO, October 12, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 (OL)
	)	(Emergency Planning-- Phase I)
(Shoreham Nuclear Power Station, Unit 1)	)	

TESTIMONY OF JACK A. NOTARO AND ROBERT L. POLTRINO
FOR THE LONG ISLAND LIGHTING COMPANY
ON PHASE I EMERGENCY PLANNING CONTENTION 13 --
INTERIM SAFETY PARAMETER DISPLAY SYSTEM

PURPOSE

This testimony demonstrates that LILCO has designed a Safety Parameter Display System (SPDS) for Shoreham that will include the recommended functional capabilities of NUREG-0696 as modified by SECY 82-111. The interim SPDS is designed for use until the permanent SPDS is available.

The interim SPDS provides for validation of data by providing redundant displays of most parameters, and has a trending capability for both radiological and non-radiological parameters. All SPDS displays and parameters (radiological and non-radiological) are available in the TSC. All radiological displays and radiological parameters are available in the EOF;

EOF personnel may communicate directly with the TSC for any additional technical parameters. The interim SPDS will provide graphic displays in the control room, to augment the information available to the operator for determining if a transient or accident has occurred, and to aid the operator in tracking the course of an accident.

ATTACHMENTS

Attachment 13-1	Professional Qualifications of Jack A. Notaro
Attachment 13-2	Professional Qualifications of Robert L. Poltrino
Attachment 13-3	SECY 82-111, "NRC Staff Recommendations on the Requirements for Emergency Response Capability"
Attachment 13-4	Interim SPDS Parameters

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INTERIM SAFETY PARAMETER DISPLAY SYSTEM

Q1. Please state your names and business addresses.

A1. My name is Jack A. Notaro; my business address is Shoreham Nuclear Power Station, P. O. Box 628, Wading River, New York, 11792.

My name is Robert L. Poltrino; my business address is Stone and Webster Engineering Corporation, 245 Summer Street, Boston, Massachusetts, 02110.

Q2. What are your present positions with your companies?

A2. [Notaro] I am the Operating Engineer at the Shoreham Nuclear Power Station for LILCO.

[Poltrino] I am the principal electrical control engineer for Stone & Webster assigned to the Shoreham project.

Q3. Please state your professional qualifications.

A3. [Notaro] My professional qualifications are attached to this testimony (Attachment 13-1). My familiarity with the issues surrounding the interim SPDS stems from my daily responsibility for safe, reliable Plant Operations. This encompasses all integrated system operations, including use of the interim SPDS.

[Poltrino] My professional qualifications are attached to this testimony (Attachment 13-2). My familiarity with the issues surrounding the interim SPDS stems from my involvement with the design and design review of the Phase I Emergency Response Facilities (ERF), including the interim SPDS. Prior to my current assignment, I was an engineer in the electrical control group on Shoreham. I have been involved with Shoreham for over eight years.

[Both witnesses sponsor the remaining answers in this testimony.]

Q4. Are you familiar with Suffolk County Contention EP 13?

A4. Yes.

Q5. What does that contention say?

A5. The contention reads as follows:

> Suffolk County contends that the interim SPDS that LILCO proposes to utilize until the installation of a permanent SPDS is deficient because it does not meet minimum requirements for such a system. Specifically, the interim SPDS does not:
>
> A. provide all required parameters [NUREG-0696 at 26];
>
> B. provide for data verification [NUREG-0696 at 24];
>
> C. provide trending capability [NUREG-0696 at 25-26];
>
> D. provide information to the TSC and EOF [NUREG-0696 at 25]; and
>
> E. provide the function of aiding the operator in the interpretation of transients and accidents, nor does it provide this function during and following all events expected to occur during the life of the plant, including earthquakes [NUREG-0696 at 27].
>
> Thus, the interim SPDS does not meet the requirements of 10 CFR §§ 50.47(b)(4)(8), and (9), 10 CFR Part 50, Appendix E, Items IV.E.2 and 8, 10 CFR Part 50, Appendix A, GDC 13, and NUREGs-0696, 0737 and 0654, Item I.

Q6. What is the thrust of the contention?

A6. Generally, Suffolk County contends that LILCO's interim SPDS does not meet the guidelines of NUREG-0696 for such a system.

Q7. Are you familiar with the regulatory requirements and guidelines cited in EP 13?

A7. Yes.

Q8. Do 10 C.F.R. §§ 50.47(B)(4),(8) and (9), 10 C.F.R. Part 50, Appendix E, Items IV.E.2 and 8, 10 C.F.R. Part 50, Appendix A, GDC 13, and NUREG-0654, Item I establish requirements for SPDS?

A8. No.

Q9. What are the current regulatory requirements and guidelines for the installation of a safety parameter display system at Shoreham, and how has LILCO implemented those requirements and guidelines?

A9. SECY 82-111, "NRC Staff Recommendations on the Requirements for Emergency Response Capability" (Attachment 13-3), outlines the current guidelines for SPDS. The original technical requirements and implementation schedules for SPDS evolved through such NRC documents as NUREG-0578, NUREG-0737 and NUREG-0696. Recognizing at the time these guidelines were issued that procurement and design of a new computer system fully meeting NUREG-0696 guidelines could not be achieved at Shoreham by fuel load, due to the late emergence of the definitive information provided in NUREG-0696,

LILCO chose to implement the SPDS for Shoreham in two phases. Phase I provides for an interim SPDS, using the existing plant process computer; Phase II, a permanent SPDS, using a new computer base system. The interim SPDS would be used only until the permanent SPDS becomes available.

Prior to the issuance by the NRC of SECY 82-111, LILCO had obtained Staff concurrence to use an interim SPDS until at least April, 1983. With the revised criteria and schedule guidelines set forth in SECY 82-111, two of the specific "requirements" listed by the County in its contention are no longer suggested by the NRC: (1) that SPDS be seismically qualified, and (2) that licensees meet certain schedule commitments that had been suggested. Implementation dates for SPDS are now to be determined by discussions between the individual licensee and the NRC project manager.

Q10. Please describe the Phase I interim SPDS.

A10. The interim SPDS is based on (1) the existing plant process computer, (2) the Radiation Monitoring System (RMS) computers, and (3) a portion of the permanent data acquisition system. This additional diagnostic information will be available in a concentrated location to the station operators and technical and management level personnel.

Q11. How does Phase II differ from Phase I?

A11. Phase II is based on a new multi-computer system consisting of four independent computers, two of which are seismically qualified. Any one of the four computers is capable of providing SPDS displays.

Q12. What "parameters" are suggested in NUREG-0696 for SPDS displays, as referred to by the County in EP 13(A)?

A12. NUREG-0696 does not suggest any specific "parameters" to be incorporated in the SPDS. NUREG-0696 does, however, state:

> The important plant <u>functions</u> related to the primary display while the plant is generating power shall include, but not be limited to:
>
> a. Reactivity control
>
> b. Reactor core cooling and heat removal from primary system
>
> c. Reactor coolant system integrity
>
> d. Radioactivity control
>
> e. Containment integrity

(Emphasis added.)

Q13. Do the interim SPDS displays for Shoreham monitor all the important plant functions suggested in NUREG-0696?

A13. Yes.

Q14. What functions are available from the interim SPDS?

A14. The interim SPDS monitors the status of the following functions:

a. Reactivity control
b. Reactor core coolant/heat removal
c. Reactor coolant system integrity
d. Containment integrity/heat removal
e. Radioactivity control

Functions (a) through (d) do have parameters that are displayed on a color CRT located on panel 1H11*PNL-603, directly in front of the operator's desk in the main control room.

The Radiation Monitoring System (RMS) providing the radioactivity control information listed in (e), above, is a computer-based radiological information system that provides the status of radiation levels throughout the plant. Data from individual radiation monitors is inputted to the computer system to provide continuous real time radiological status for the plant. The computer system also provides trending capability and automatic report generation. Set points are provided for each radiation monitor; these set points will alarm to alert plant personnel of unusual conditions. The RMS system provides the station operators with parameters that monitor the status of radioactivity control.

Thus, the interim SPDS for Shoreham addresses each of the five categories referenced in NUREG-0696.

Q15. What specific parameters does the interim SPDS monitor?

A15. For a complete list of the specific parameters monitored by Shoreham's interim SPDS, see Attachment 13-4, attached to this testimony.

Q16. In response to EP 13(B), does the interim SPDS provide for data verification?

A16. Yes. Data validation is the process by which the accuracy of the data being displayed is determined. Many of the interim SPDS parameters displayed or available for display have at least two redundant channels for every parameter used for the interim SPDS. Therefore, the operator can readily validate data presented by the interim SPDS by comparing redundant parameter channels. The operator would be aware of any discrepancy between redundant channels, and could refer to other SPDS parameters or board-mounted instrumentation to resolve any conflicting readings.

Q17. In response to EP 13(C), does the interim SPDS provide a trending capability?

A17. Yes, the interim SPDS has the capability to provide

analyses of trends. The RMS has historical logs and graphic trend displays, both of which can provide trend information for a radiological parameter or group of parameters. These printouts and displays are available at the RMS printer and the CRT in the control room. For the remainder of the SPDS, the operator has available in the control room a two-pen trend recorder. Any analog point in the process computer data base can be assigned to this recorder to provide trending information. (This recorder is located on panel 1H11*PNL-602.) In addition, there is available to the control room operator eight special logs to which the operator can assign any analog computer point in the process computer data base along with the desired scan period for historical logging and trend analysis.

Q18. In response to EP 13(D), does the interim SPDS provide information to the Technical Support Center (TSC) and the Emergency Operations Facility (EOF)?

A18. Yes. The SPDS parameters and, in fact, the entire process computer data base, is available from printers located in the TSC. In addition, the CRT in the TSC may be utilized to display individual parameters from the process computer data base or to display any of the graphics available from the plant process computer. The

entire contents of the data base of the RMS -- both CRT graphics and hard copy printouts -- are available at the TSC.

Only radiological parameters are provided at the EOF. It is this radiological information which is of prime importance at the EOF. Any additional information of interest to the personnel in the EOF is available via direct communication to the TSC.

SECY 82-111 deleted the suggested guideline that SPDS displays be included in the TSC and EOF.

Q19. In response to EP 13(E), does the interim SPDS aid the operator in the interpretation of transients and accidents?

A19. Yes. The SPDS displays provide the station operator with additional diagnostic information in a concentrated location, aiding the operator in tracking the course of an accident.

Q20. Will the interim SPDS be available following all events expected to occur during the life of the plant, including earthquakes?

A20. The plant process computer is powered from a seismically qualified inverter, which, in turn, is fed from a

safety-grade diesel generator and battery. The process computer is located in a room with a redundant safety-grade ventilation system in a non-harsh environment. With the exception of seismic events, the interim SPDS will function during any event expected to occur during the life of the plant. The plant process computer was not designed to be seismically qualified. SECY 82-111 has deleted all seismic provisions for the SPDS. However, the permanent SPDS will have two seismically qualified CRT's in the control room, driven by two redundant seismically qualified computers.

Q21. Please summarize your testimony.

A21. LILCO has designed a Safety Parameter Display System (SPDS) for Shoreham that will include the recommended functional capabilities of NUREG-0696 as modified by SECY 82-111. The interim SPDS is designed for use until the permanent SPDS is available.

The interim SPDS provides for validation of data by providing redundant displays of most parameters, and has a trending capability for both radiological and non-radiological parameters.

All SPDS displays and parameters (radiological and non-radiological) are available in the TSC. All

radiological displays and radiological parameters are available in the EOF; EOF personnel may communicate directly with the TSC for any additional technical parameters. The interim SPDS will provide graphic displays in the control room, to augment the information available to the operator for determining if a transient or accident has occurred, and to aid the operator in tracking the course of an accident.

PROFESSIONAL QUALIFICATIONS

JACK A. NOTARO

Operating Engineer

LONG ISLAND LIGHTING COMPANY

My name is Jack A. Notaro. My business address is Long Island Lighting Company, Shoreham Nuclear Power Station, P.O. Box 628, Wading River, New York, 11792. I am employed by Long Island Lighting Company as an Operating Engineer and have held this position since July 1978. I am assigned as the Operating Engineer at the Shoreham Nuclear Power Station. In this capacity, I am responsible for the development and implementation of the Station's operational activities including the direct n of startup, day-to-day operation and shutdown of station equipment. My responsibilities also include the implementation of initial, requalification and replacement training programs for licensed and unlicensed operators, and the development, review and implementation of the operations section of the Station Operating Manual.

I received a Bachelor of Mechanical Engineering degree from City College of New York in 1970; and a Master of Business Administration from Adelphi University in 1974. In July 1976, I completed the General Electric Boiling Water Reactor Simulator Program and obtained certification as a Senior

Reactor Operator. I have completed several industry seminars and training programs including: BWR Design Orientation, BWR Technology, BWR Observation Training, Nuclear Power Plant Technology, Radiation Protection, Basic Health Physics, Vibration Analysis, Statics, Strength of Materials & Dynamics, Management of Maintenance Storekeeping & Inventories, QA for the Nuclear Industry, Inservice Inspection & QA During Operations, Basic Radiography, Magnetic Particle & Liquid Penetrant Testing, Basic Ultrasonics, Nuclear Power QA, Inservice Inspection Symposium, Operations Quality Assurance, Fire Fighting Training, Limerick Simulator Capability, and Simulator Refresher Training.

I joined Long Island Lighting Company in June 1970 and was assigned to the Maintenance Section at the Northport Power Station. In January 1972 I was transferred to the Electric Production Department and my assigned duties included maintenance scheduling, manpower allocation, equipment testing and station performance analysis.

In March 1973 I was assigned to the Quality Assurance Section at the Shoreham Nuclear Power Station. I was subsequently promoted to Station Operating Quality Assurance Engineer in July 1974. My duties included initial development and implementation of the operational quality assurance program. This program included reviews, audits, surveillances, inspections, selection and training of personnel, development

of procedures and instructions, and the utilization of consultants and contractors. In addition, I was responsible for licensing and inspection activities associated with the U.S. Nuclear Regulatory Commission.

In August 1978 I was assigned to the Vermont Yankee Nuclear Power Station to observe startup of the unit following a refueling outage. I witnessed the completion of the integrated leak rate test, the primary systems hydrostatic pressure test, and the drywell inspection. I observed approach to criticality, criticality, plant heat-up and transfer to run. I also witnessed a half-scram recovery during plant heat-up.

From March 1981 to May 1981 I was assigned to the Operations Section of the Millstone Nuclear Power Station for the completion of the Unit 1 refueling outage. I participated in all significant pre- and post-refueling outage surveillance testing and inspections, and I actually took part in refuel bridge operations including control rod removal and replacement, channeled and dechanneled fuel movements, core inspections and verifications, dropped fuel bundle evaluations, and recovery.

I was assigned to the Operations Section of the Millstone Nuclear Power Station in June 1981. I participated in routine BOP and NSSS system surveillance testing, and high risk I&C and operations equipment and system surveillance testing. I conducted heat balances, core flow calculations and

subsequent nuclear instrumentation calibrations with, and without, the main computer available. In addition, I witnessed implementation of emergency notification procedures, as well as manipulated controls for power downs, return to power, Tech Spec LCO's, control rod repositioning, and stuck control rod surveillance testing. I also participated in half scram and full scram recoveries and the subsequent investigations, evaluations and notifications.

I am a member of the American Society for Quality Control and past member of the Edison Electric Institute - Quality Assurance Task Force (EEI-QATF) and the EEI-QATF Operations Subcommittee.

PROFESSIONAL QUALIFICATIONS

ROBERT L. POLTRINO

Control Engineer, Control Systems Division

STONE & WEBSTER ENGINEERING CORPORATION

My name is Robert Poltrino. My business address is 245 Summer Street, Boston, Massachusetts 02110. I am employed by Stone & Webster Engineering Corporation as the Principal Electrical Control Engineer on the Shoreham Project and have held this position since September 1980. In this capacity, I have overall responsbility for the design and layout of auxiliary control and relay panels, preparation of the purchase specifications for those panels, review and approval of nuclear steam supply system (NSSS) control panel layout drawings, and the preparation of elementary control diagrams.

I was awarded a Bachelor of Science degree in electrical engineering from Northeastern Univeristy. Since degree conferral, I have successfully completed all but ten credit hours towards the Master of Science degree in electrical engineering at Northeastern.

Prior to joining Stone & Webster, I served in the United States Army from July 1969 - July 1973. As a Project Officer in the U.S. Army Training Device Agency, I was

responsible for preparation of conceptual system design and descriptions for computer-aided training systems. Prior to this, I was involved in the installation and management of high capacity microwave and tropospheric scatter communications systems and associated telecommunications switching centers. These systems were installed at some 20 different sites throughout Southeast Asia. I also monitored and evaluated contractor operations and performance overseas.

Upon joining Stone & Webster Engineering Corporation (November 1973) as an Engineer, I was assigned to the electrical control group on the Long Island Lighting Company (LILCO) Shoreham Nuclear Power Station (SNPS) Unit 1. My activities included review of GE NSSS control board layout, review of vendor elementary drawings, preparation of purchase specifications for the auxiliary control and relay panels, design and layout of auxiliary control panels and benchboards, design and layout of start-up transient monitoring system interface, and the preparation of elementary controls diagrams.

Since September 1980 to the present, I have functioned as Principal Electrical Control Engineer on the Shoreham Project. Since then I have been scheduling and assigning tasks, and establishing completion dates within the electrical control group. I supervise assigned personnel, and review and

approve engineering drawings and documents prepared and issued within the control systems area of responsibility.

I am a Registered Professional Engineer in the Commonwealth of Massachusetts and the State of California.

SECY 82-111

NRC STAFF RECOMMENDATIONS
ON THE
REQUIREMENTS FOR
EMERGENCY RESPONSE CAPABILITY

March 10, 1982

CONTENTS

CONTENTS (Continued)

EMERGENCY RESPONSE CAPABILITY

1. INTRODUCTION

This report was prepared as a result of a review by the Committee to Review Generic Requirements (CRGR). The recommendations herein have been developed by the program offices and are supported by CRGR. The report represents the staff's attempt to distill the fundamental requirements for nuclear plant Emergency Response Capability from the wide range of guidance documents that NRC has issued. It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be ignored; they are still useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document.

These fundamental requirements are further specification of the general guidance specified previously by the Commission in its regulations, orders and policy statements on emergency planning and TMI issues. It is intended that these fundamental requirements would be applicable to licensees of operating nuclear power plants and holders of construction permits for 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. In this regard, it is expected that the staff would review CP and ML applications against the guidance in the current Standard Review Plan, and this might lead to more detailed requirements than prescribed in this document.

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, the staff proposes that plant-specific schedules be established which take into account the unique status of each plant. The following sequence for developing implementation schedules is proposed.

When the basic requirements for emergency response capabilities and facilities are finalized, they should be transmitted to licensees by a generic letter from NRR, promulgated to NRC staff, and incorporated as regulatory requirements (e.g., in the Standard Review Plan or by regulation or Order, as appropriate). The letter to licensees should request that licensees submit a proposed schedule for completing actions to comply with the basic requirements. Each licensee's proposed schedules would then be reviewed by the assigned NRC Project Manager, who would discuss the subject with the licensee and mutually agree on schedules and completion dates. The implementation dates would then be formalized into an enforceable document.

The basic 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 upgraded 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 basic requirements. The NRC staff does not recommend that existing guidance 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 basic requirements and guidance.

The following sections describe NRC staff recommendations on basic requirements, their interrelationships, and NRC actions to improve management of emergency response regulation. Reference documents are cited with a description of content as it relates to specific initiatives.

2. USE OF EXISTING DOCUMENTATION

The NRC staff recommends that 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
0801	-	Evaluation Criteria for 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

Regulatory Guides		
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

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 symptom 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. However, installation of the SPDS should not be delayed by slower progress on other initiatives. The SPDS should not be contingent on completion of the control room design review. NRC does not plan to impose additional requirements on licensees regarding SPDS.

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. 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.

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

   a. The SPDS is an improvement in the control room 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 but need not be centralized.

5. Specific implementation plans and reasonable, achievable schedules should be established by agreement between the NRC Project Manager and each individual licensee. The NRC office responsible for implementing each requirement should 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.

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.

7. 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 Office Director.

8. 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 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 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)

Current Regulatory Requirements

No licensee action is required.

Functional Statement

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.

Recommended Requirements

1. 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.

2. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) forms the basic safety components required 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. After the SPDS has been installed, operating procedures should be available that will allow timely and correct safety status assessment when the SPDS is not available.

3. 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 judgment of individual plant licensees, taking into account the importance of prompt implementation.

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

5. Minimum information to be provided shall be sufficient to provide information to plant operators about:

   a. Reactivity control
   b. Reactor core cooling and heat removal from the primary system
   c. Reactor coolant system integrity
   d. Radioactivity control
   e. Containment conditions

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

6. 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.

7. The licensee's proposed implementation of an SPDS system shall be reviewed in accordance with the licensee's technical specifications to determine whether the changes involve an unreviewed safety question or change of technical specifications. If they do, they 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. However, the licensee's analysis shall be submitted to NRC promptly on completion of review by the licensee's offsite 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.

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 symptom-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 should be accepted by NRC.

Reference Documents

NUREG-0660 -- Need for SPDS identified

March 11, 1982

SECY-82-111

POLICY ISSUE
(Notation Vote)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

Purpose: To request Commission approval of a set of basic requirements for emergency response capability and approval for the staff to work with licensees to develop plant-specific implementation schedules.

Discussion: One of the first issues reviewed by the Committee to Review Generic Requirements (CRGR) was the broad area of emergency response facilities and capabilities at nuclear plants. The Committee found that implementation schedules were not being coordinated within the NRC. In addition, existing NRC documents published as guidance to licensees were sometimes being used as firm requirements. Discussions with industry representatives and the staff indicated that some licensees had slowed down on work in this area pending NRC clarification of its requirements. Some utilities have virtually stopped work on some of the items, while others have proceeded and, in some cases, completed some of the items. The Committee recommended that steps be taken by the Office Directors involved to clarify the requirements and implementation schedules for the Safety Parameter Display System (SPDS), Control Room Design Review, upgraded Emergency Operating Procedures, Regulatory Guide 1.97, Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). In my memo to the Commission dated December 31, 1981, I noted that the DEDROGR staff would work with the program offices to clarify the basic requirements in this area and establish a revised implementation plan.

Enclosed are the staff's recommendations for the requirements in the broad area of emergency response facilities and capabilities outlined above. The requirements were developed by the program offices

Contact:
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49-29704

and are supported by CRGR. The enclosure represents a distillation of fundamental requirements from the broad range of guidance documents that NRC has issued (principally NUREG reports and Regulatory Guides). The staff intends that the guidance documents referred to in the enclosure not be used to impose requirements on licensees, but rather that they be used as sources of guidance for NRC reviewers and licensees regarding acceptable means for meeting the fundamental requirements proposed.

In discussions with owners' groups and individual licensees, the staff has learned that the Commission approved schedule of October 1, 1982, for implementation of the TSC and EOF probably cannot be met. In recognition of this fact and the difficulty of implementing generic deadlines, the staff is proposing that plant-specific schedules be established which take into account the unique status of each plant. Each licensee would be requested to submit a proposed schedule for completing the actions to comply with the fundamental requirements. The NRC Project Manager for each plant should be knowledgeable of the overall work effort going on at a plant and, based on guidance received from NRC management, could reach agreement with licensees on schedules which optimize use of utility and NRC resources. The agreed upon completion dates would be formalized in an order. By this approach, future staff coordination problems regarding implementation schedules will be avoided.

<u>Resource Estimates:</u> The costs to licensees to implement the requirements proposed in the enclosure were included in the estimates set out in NUREG-0660.

<u>Recommendation:</u> That the Commission:

1. <u>Approve</u> the fundamental requirements described in the enclosure.

2. <u>Approve</u> the issuance of the requirements in the enclosure by 50.54f letters as a revision to NUREG 0737.

3. <u>Approve</u> the method for establishing plant-specific implementation schedules described in the enclosure.

4. Approve the implementation of these requirements through plant-specific orders.

5. Note that the staff intends to use the previously issued NUREG reports and Regulatory Guides as guidance documents only.

Scheduling: Licensees are currently required to establish a TSC and EOF by October 1. Prompt action on this paper is required in order to provide guidance to licensees.

William J. Dircks
Executive Director for Operations

Enclosure:
NRC Staff Recommendation on the Requirements for Emergency Response Capability

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Monday, March 29, 1982.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, March 22, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ENCLOSURE

EMERGENCY RESPONSE CAPABILITY

COMMISSION BRIEFING

VICTOR STELLO, JR.

APRIL 15, 1982

COMMISSION DECISIONS RECOMMENDED

1. APPROVAL OF STAFF'S PROPOSED SET OF BASIC REQUIREMENTS WHICH HAVE BEEN DISTILLED FROM THE BROAD RANGE OF GUIDANCE DOCUMENTS THAT NRC HAS ISSUED.

2. APPROVAL OF STAFF'S PROPOSED IMPLEMENTATION PLAN

-- ISSUE PROPOSED REQUIREMENTS TO LICENSEES BY 50.54f LETTERS AS A REVISION TO NUREG-0737

-- ESTABLISH PLANT-SPECIFIC SCHEDULES BY MUTUAL AGREEMENT WITH LICENSEES

-- FORMALLY IMPLEMENT REQUIREMENTS AND SCHEDULES THROUGH PLANT-SPECIFIC ORDERS

BACKGROUND

CRGR REVIEW OF OVERALL NRC ACTIVITIES IN EMERGENCY RESPONSE (12/3/81)

-- NRC OFFICE ACTIVITIES NEEDED BETTER COORDINATION

-- ACTUAL REQUIREMENTS NOT CLEAR

-- NUREGS AND REG GUIDES SOMETIMES USED BY STAFF AS FIRM REQUIREMENTS

INITIAL DRAFT OF STAFF RECOMMENDATIONS (12/29/81)

ACRS BRIEFING (1/8/82)

FINAL STAFF RECOMMENDATIONS CONSIDERING ALL COMMENTS (3/10/82)

S C O P E

EMERGENCY RESPONSE FACILITIES

-- TECHNICAL SUPPORT CENTER (TSC)

-- OPERATIONAL SUPPORT CENTER (OSC)

-- EMERGENCY OPERATIONS FACILITY (EOF)

CONTROL ROOM IMPROVEMENTS

-- SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

-- CONTROL ROOM DESIGN REVIEW

-- INSTRUMENTS FOR ACCIDENTS (AS IDENTIFIED IN REG GUIDE 1.97)

OPERATOR CAPABILITY

-- IMPROVED EMERGENCY OPERATING PROCEDURES

-- TRAINING

PROPOSED BASIC REQUIREMENTS - INTEGRATION OF ACTIVITIES

IT IS <u>ESSENTIAL</u> THAT THE ACTIVITIES ON CONTROL ROOM IMPROVEMENTS, UPGRADING OF EMERGENCY OPERATING PROCEDURES, AND STAFF TRAINING BE INTEGRATED AT EACH PLANT. THE TSC AND EOF DESIGN AND IMPLEMENTATION ARE RELATED TO CONTROL ROOM IMPROVEMENTS IN COMMUNICATIONS AND INSTRUMENTATION.

PROPOSED BASIC REQUIREMENTS - SPDS

-- PROMPT IMPLEMENTATION OF SPDS IS IMPORTANT AS A SAFETY IMPROVEMENT, PROVIDING INFORMATION, AS A MINIMUM, ON:

- REACTIVITY CONTROL
- REACTOR CORE COOLING AND HEAT REMOVAL
- REACTOR COOLANT SYSTEM INTEGRITY
- RADIOACTIVITY CONTROL
- CONTAINMENT CONDITIONS

-- SHOULD BE DESIGNED ACCORDING TO GOOD HUMAN FACTORS PRINCIPLES

-- MUST BE CONSIDERED DURING CONTROL ROOM DESIGN REVIEW

-- NO SEISMIC, CLASS 1E OR SINGLE-FAILURE REQUIREMENTS

-- POST-IMPLEMENTATION REVIEW

PROPOSED BASIC REQUIREMENTS - CONTROL ROOM DESIGN REVIEW

-- PURPOSE IS TO IDENTIFY HUMAN ENGINEERING DISCREPANCIES

-- MULTIDISCIPLINARY REVIEW TEAM

-- FUNCTION AND TASK ANALYSIS REQUIRED

-- SIGNIFICANT DISCREPANCIES SHOULD BE CORRECTED

-- MUST BE INTEGRATED WITH OTHER ACTIVITIES SUCH AS SPDS, UPGRADED EMERGENCY OPERATING PROCEDURES, OPERATOR TRAINING AND NEW REG GUIDE INSTRUMENTATION

-- NRR REVIEW ON AN AUDIT BASIS

PROPOSED BASIC REQUIREMENTS - REG GUIDE 1.97

FOR CONTROL ROOM - INDICATION OF TYPE A,B,C,D,E VARIABLES

- INDICATION OF WIND DIRECTION, SPEED AND ATMOSPHERIC STABILITY
- EQUIPMENT QUALIFICATION AS DETERMINED BY PENDING RULEMAKING

FOR TSC - INDICATION OF TYPE A,B,C,D,E VARIABLES NEEDED FOR TSC FUNCTION

- NEED NOT MEET CLASS 1E, SINGLE-FAILURE OR SEISMIC QUALIFICATION REQUIREMENTS

FOR EOF - INDICATION OF VARIABLES NECESSARY TO PERFORM EOF FUNCTION

- NEED NOT MEET CLASS 1E, SINGLE-FAILURE OR SEISMIC QUALIFICATION REQUIREMENTS

PROPOSED BASIC REQUIREMENTS - EMERGENCY OPERATING PROCEDURES

-- REANALYZE ACCIDENTS AND PREPARE TECHNICAL GUIDELINES

-- UPGRADE EMERGENCY OPERATING PROCEDURES CONSISTENT WITH GUIDELINES

-- OPERATOR TRAINING ON PROCEDURES PRIOR TO IMPLEMENTATION

-- MUST BE INTEGRATED WITH SPDS AND CONTROL ROOM DESIGN REVIEW

-- NRC REVIEW AND APPROVAL OF TECHNICAL GUIDELINES

-- OPPORTUNITY FOR PREIMPLEMENTATION REVIEW OF PROCEDURES

PROPOSED BASIC REQUIREMENTS - EMERGENCY RESPONSE FACILITIES

-- GENERAL REQUIREMENTS SPECIFIED FOR:
- LOCATION
- SIZE
- RADIATION PROTECTION
- RECORDS
- EQUIPMENT
- COMMUNICATIONS
- STAFFING

-- NO NRC APPROVAL OF CONCEPTUAL DESIGNS REQUIRED BEFORE CONSTRUCTION

PROPOSED IMPLEMENTATION PLAN

-- ISSUE PROPOSED BASIC REQUIREMENTS TO LICENSEES BY 50.54f LETTERS AS A REVISION TO NUREG 0737

-- NRC PROJECT MANAGERS (WITH MANAGEMENT GUIDANCE) NEGOTIATE PLANT-SPECIFIC SCHEDULES WITH LICENSEES

-- NRR IMPLEMENT FORMAL REQUIREMENTS THROUGH PLANT-SPECIFIC ORDERS

-- CURRENT OCTOBER 1982 GENERIC DEADLINE FOR OPERATIONAL EMERGENCY RESPONSE FACILITIES WILL BE EXTENDED ON CASE-BY-CASE BASIS

-- DESIGN AND INSTALLATION OF INPLANT SYSTEMS AND FACILITIES SHOULD BE EXPEDITED

<u>STAFF USE OF NUREGS AND REG GUIDES</u>

-- TO BE USED AS GUIDANCE ONLY, NOT REQUIREMENTS

-- EDO INSTRUCTIONS TO STAFF

-- DISCLAIMER STATEMENT IN NUREGS

NUREG-0737	--	Specified SPDS
NUREG-0696	--	Functional criteria for SPDS
NUREG-0835	--	Specific acceptance criteria keyed to 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

Current Regulatory Requirements

As specified in Item I.D.1 in NUREG-0737, the implementation schedule is still to be developed.

Functional Statement

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.

Recommended Requirements

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

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

   b. The use of function and task analysis (that had been used as the basis for developing emergency operating procedure Technical Guidelines) 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.

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

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

2. 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.

3. 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.

Documentation and NRC Review

1. 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. NRC approval is not required before licensees conduct their reviews.

2. Selected licensees will undergo an in-progress audit by the NRR human factors staff based on the program plans and advice from resident inspectors.

3. All licensees shall submit a summary report outlining proposed control room changes. The report will also provide a summary justification for human engineering discrepancies with safety significance to be left uncorrected or partially corrected.

4. Within two weeks after receipt of the licensee's summary report, the NRC will inform the licensee whether it will conduct a pre-implementation onsite audit. The decision will be based on the content of the program plan, summary report, and 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 2.

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

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

   b. 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 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).

6. 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 5 above, except that it may or may 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.

7. 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 dates after June 1983, Control Room Design Review will be required prior to licensing.

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 symptom-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 should be accepted by NRC.

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	--	October 1981 draft for comment; staff evaluation criteria.

REGULATORY GUIDE 1.97

6. APPLICATION TO EMERGENCY RESPONSE FACILITIES

Current Regulatory Requirements

No licensee action is required.

Functional Statement

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

Recommended Requirements

1. 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 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.

2. Technical Support Center (TSC)

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

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

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

3. Emergency Operations Facility (EOF)

a. Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be provided in the EOF.

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

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) redundancy 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)

Current Regulatory Requirements

NUREG-0737, Item I.C.1, which has been approved by the Commission for implementation.

Functional Statement

Symptom-based 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.

Recommended Requirements

1. 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.

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

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

4. Implement upgraded EOPs.

Documentation and NRC Review

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

2. 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:

   a. 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.

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

c. A description of the program for validation of the EOPs.

d. A brief description of the training program for the upgraded EOPs.

3. All procedures generation packages will be reviewed. 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 ugpraded EOPs will be reviewed for technical adequacy in conjunction with the NRC Reactor Inspection Program.

Reference Documents

NUREG-0660, Item I.C.1, I.C.8, I.C.9

NUREG-0799

8. EMERGENCY RESPONSE FACILITIES

Current Regulatory Requirements

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. Provision for communications with Federal emergency response organizations. Such communications 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. Provision for communications 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.

Technical Support Center (TSC)

Functional Statement

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 predesignated 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.

Recommended Requirements

The TSC will be:

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

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

3. Structurally built in accordance with the National Uniform Building Code.

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

5. 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.

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

7. 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:

   (a) 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

   (b) 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.

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

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

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

Operational Support Center (OSC)

Functional Statement

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

Recommended Requirements

The OSC will be:

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

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

Emergency Operations Facility (EOF)

Functional Statement

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, determination of recommended 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 radioactive releases from the plant and to determine dose projections.

Recommended Requirements

The EOF will be:

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

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

3. Structurally built in accordance with the National Uniform Building Code.

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

5. 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.

6. Capable of reliable collection, storage, analysis, displays 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:

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

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

7. 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.

8. Staffed in accordance with Table 2 (previous guidance approved by the Commission). Reasonable exceptions to the 30-minute and 1-hour time limits for staffing should be justified and will be considered by NRC staff.

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

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

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 "recommended requirements" which should be satisfied. These recommended 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 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 basic 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.

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
and implementation schedule.

Eisenhut letter to power reactor licensees 4/25/80 -- 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
A. 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.
B. 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

*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

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 (Co[illegible])

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 (Co.   d)

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 pers nnel	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. Director 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.

Attachment 13 - 4

CHART

04/29/82

S.P.D.S. MAIN MENU - GRAPHIC NO. 10 -TIME- HHMMSS

REACTIVITY CONTROL

O NO. 11 - APRM LEVELS

\- NO. 12 - CONTROL ROD POSITIONS

REACTOR CORE COOLANT / HEAT REMOVAL

O NO. 13 - RX DOME PRESS, WR/FZ LEVELS, REF. LEG DW TEMP.

O NO. 14 - CORE SPRAY FLOW, SUPP POOL TEMPERATURE

REACTOR COOLANT SYSTEM INTEGRITY

O NO. 15 - RX DOME PRESS. DW PRESS, SUPP POOL PRESS

O NO. 16 - DW AVG TEMP, SUPP POOL WATER LEVELS

O NO. 17 - ADS/SRV POSITIONS, WR/FZ LEVELS

CONTAINMENT INTEGRITY / HEAT REMOVAL

O NO. 18 - RX BLDG FLOOD LEVEL: RX BLDG DELTA PRESS.
DW PRESS, SUPP POOL PRESS, DW AVG TEMP.

O NO. 19 - RX DOME PRESS, SUPP POOL LEVEL, SUPP POOL TEMP.

O NO. 20 - DW / SUPP POOL H2 CONCENTRATION

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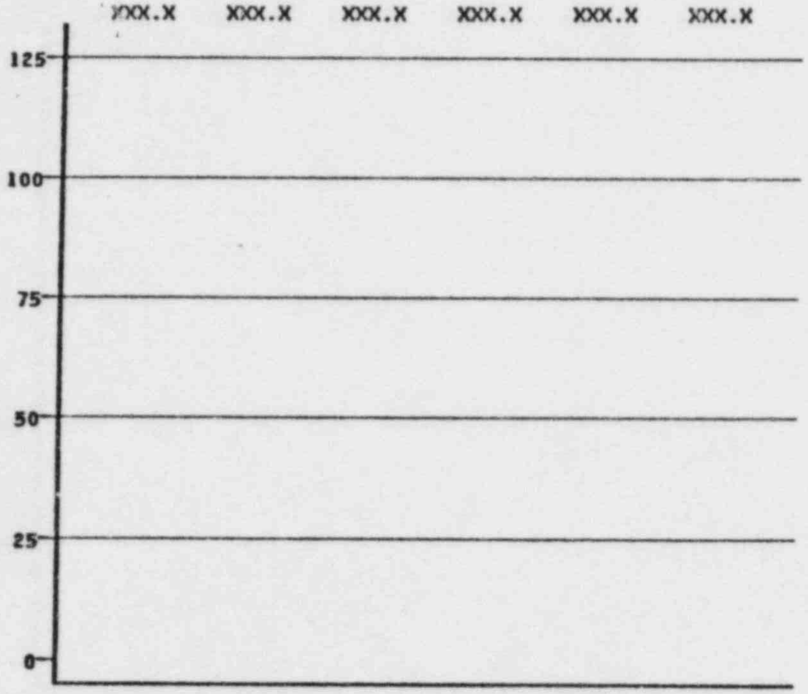
REACTIVITY CONTROL - GRAPHIC NO. 11

-TIME-
HHMMSS

	APRM A	APRM B	APRM C	APRM D	APRM E	APRM F
	B000	B001	B002	B003	B004	B005
APRM BYPASSED	A536	A537	A538	A539	A540	A541
	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X



B000 APRM A FLUX LEVEL
B001 APRM B FLUX LEVEL
B002 APRM C FLUX LEVEL
B003 APRM D FLUX LEVEL
B004 APRM E FLUX LEVEL
B005 APRM F FLUX LEVEL

CHART

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CONTROL ROD POSITIONS - GRAPHIC NO. 12

```
 YY |
 51 |
 47 |
    |         0       0       0       0       0
 43 |
 39 |
    |   0     0       0       0       0       0
 35 |
 31 |
    |   0     0       0       0       0       0
 27 |
 23 |
    |   0     0       0       0       0       0
 19 |
 15 |
    |         0       0       0       0       0
 11 |
 07 |
    |                 0       0       0
 03 |
    |_____________________________________________________
      02  06  10  14  18  22  26  30  34  38  42  46  50  XX
```

GRN - FULL IN - 00   * * - UNKNOWN POSITION

YEL - DEEP - 02 THRU 28   SELECTED ROD WILL BLINK

CYN - SHALLOW - 30 THRU 46

RED - FULL OUT - 48

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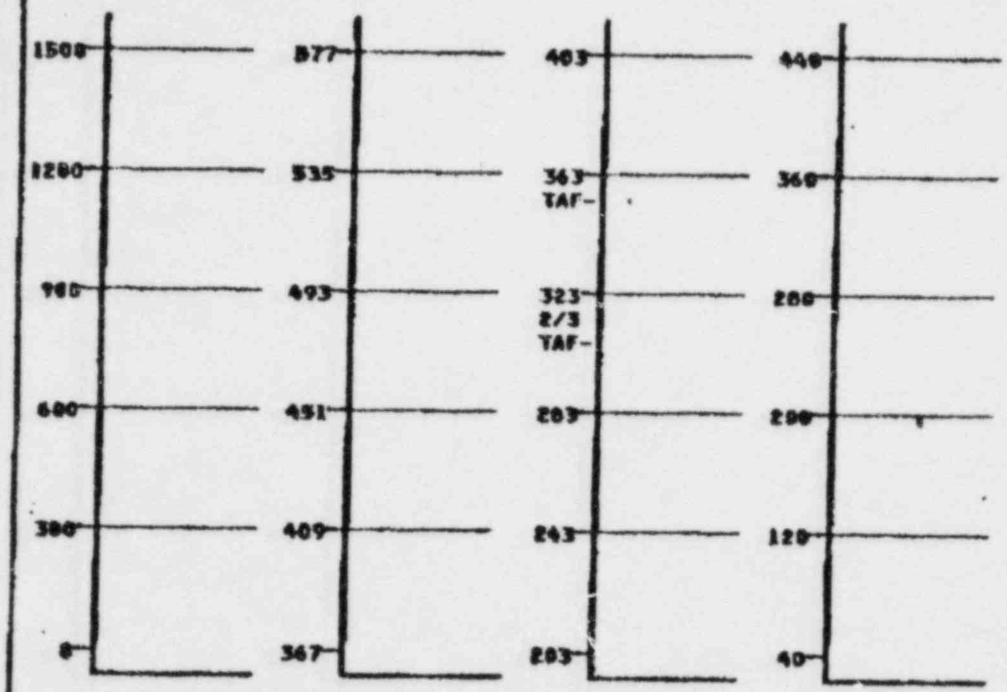
REACTOR CORE COOLANT / HEAT REMOVAL - GRAPHIC NO. 13 -TIME- [illegible]

RX PRESSURE		RX WATER LEVEL (WR)		RX WATER LEVEL (FZ)		DRYWELL TEMPERATURE	
A	B	A	B	A	B	A	B
L940	L941	L943	L944	E500	E501	L993	L994
XXXX	XXXX	XXXX	XXXX	XXXX	XXXX	XXXX	XXXX



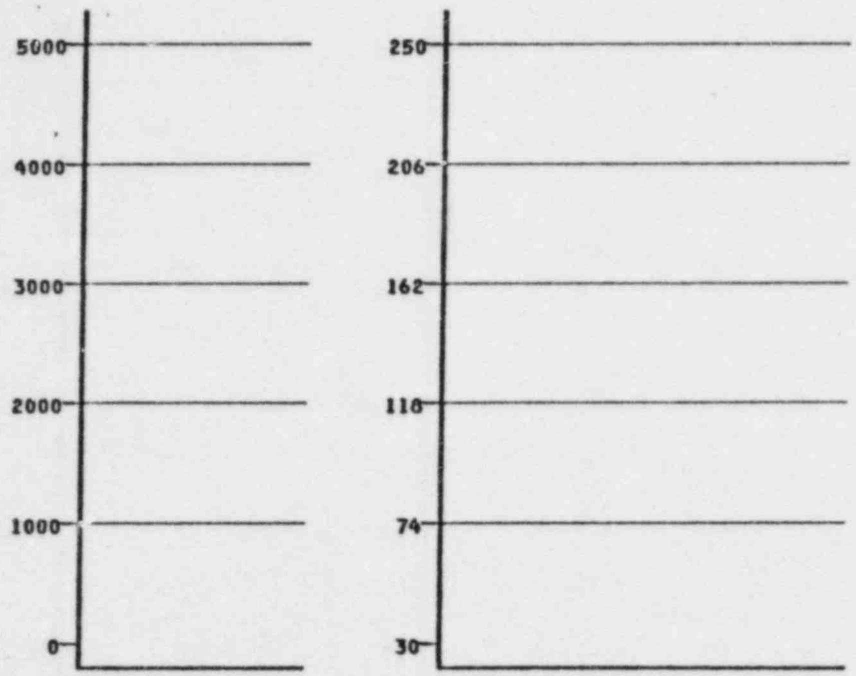
L940 REACTOR PRESSURE A PSIG
L941 REACTOR PRESSURE B PSIG
L943 REACTOR LEVEL A WR INCH
L944 REACTOR LEVEL B WR INCH

E500 REACTOR LEVEL A FZ INCH
E501 REACTOL LEVEL B FZ INCH
L993 DW T EL 145 AZ 55 DEGF
L994 DW T EL 145 AZ 230 DEGF

CHART 04/29/82

REACTOR CORE COOLANT / HEAT REMOVAL - GRAPHIC NO. 14 -TIME- HHMMSS

CORE SPRAY FLOW		SUPPRESSION POOL WATER TEMP			
A	B	1	2	3	4
G714	G718	L625	L626	L925	L926
XXXXX	XXXXX	XXXX	XXXX	XXXX	XXXX



G714 CORE SPRAY SYSTEM A GPM

G718 CORE SPRAY SYSTEM B GPM

L625 SP QUAD1 2FT WTR T DEGF

L626 SP QUAD2 2FT WTR T DEGF

L925 SP QUAD3 2FT WTR T DEGF

L926 SP QUAD4 2FT WTR T DEGF

CHART

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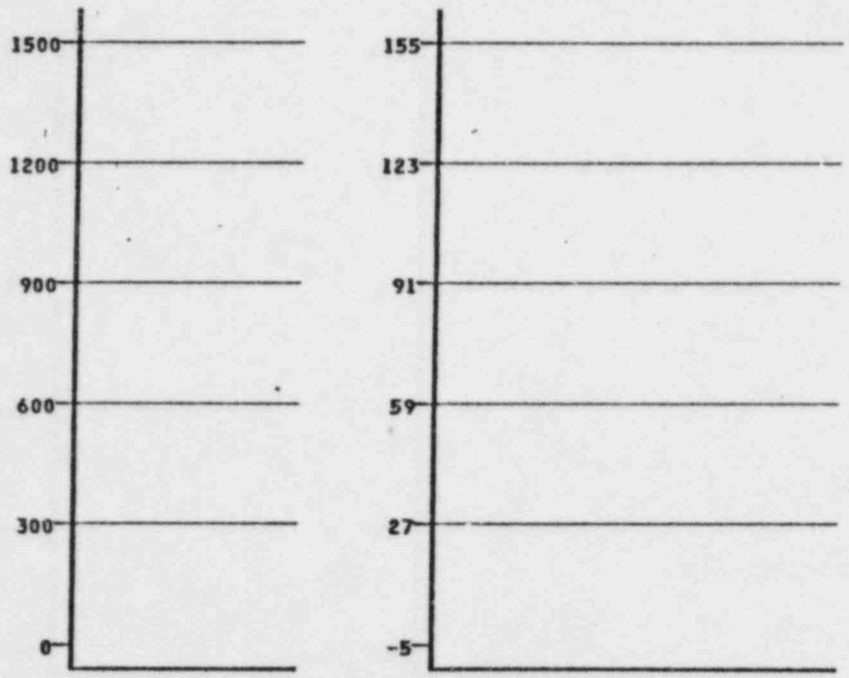
REACTOR COOLANT SYSTEM INTEGRITY - GRAPHIC NO. 15 -TIME- HHMMSS

RX PRESSURE		DRYWELL PRESSURE		SP PRESSURE	
A	B	A	B	A	B
L940	L941	F655	F656	F657	F658
XXXX.X	XXXX.X	XXX.X	XXX.X	XXX.X	XXX.X



L940 REACTOR PRESSURE A PSIG

L941 REACTOR PRESSURE B PSIG

F655 DRYWELL WR P A PSIG

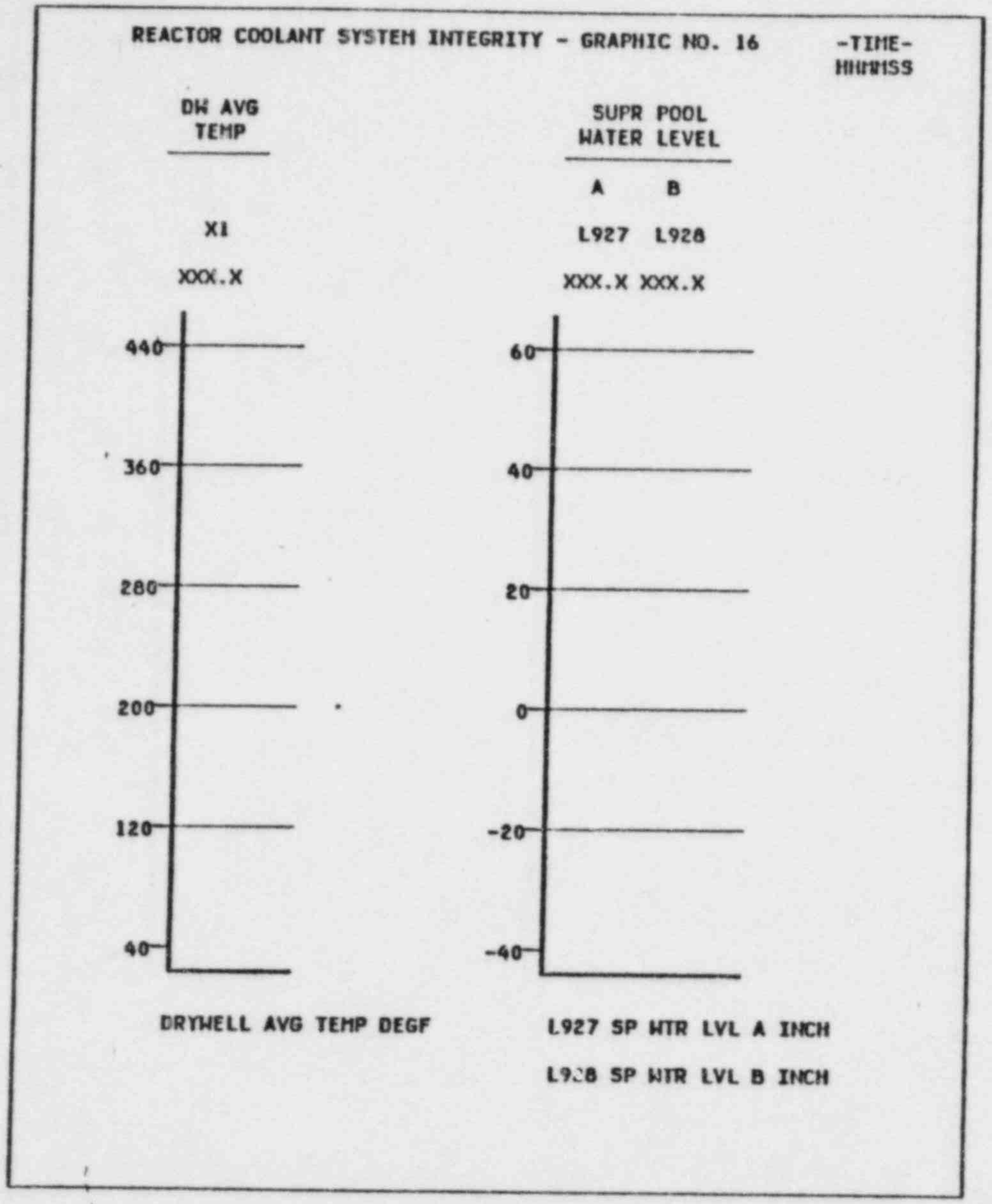
F656 DRYWELL WR P B PSIG

F657 SUPR CHMBR WR P A PSIG

F658 SUPR CHMBR WR P B PSIG

CHART

04/29/82



CHART

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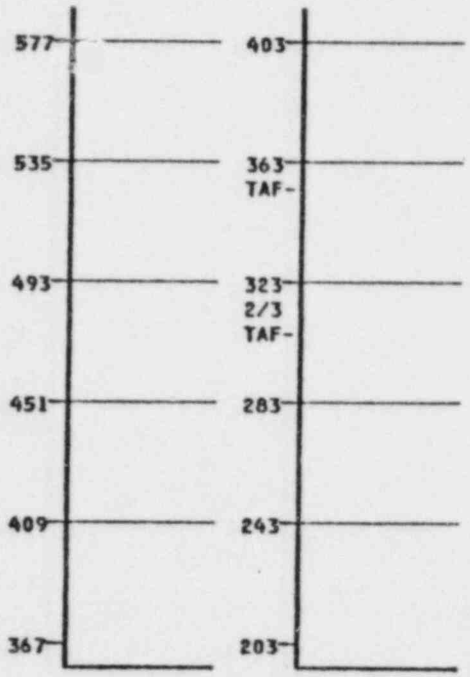
REACTOR COOLANT SYSTEM INTEGRITY - GRAPHIC NO. 17 -TIME- HHMMSS

ADS / SRV TAIL PRESS
OPEN / CLOSED

ADS/SRV	POSITION	PSIG
E507 A	XXXX	XXX.X
E511 B	XXXX	XXX.X
E514 C	XXXX	XXX.X
E515 D	XXXX	XXX.X
E520 E	XXXX	XXX.X
E522 F	XXXX	XXX.X
L947 G	XXXX	XXX.X
L948 H	XXXX	XXX.X
E527 J	XXXX	XXX.X
E528 K	XXXX	XXX.X
G703 L	XXXX	XXX.X

RX WATER LEVEL (WR)		RX WATER LEVEL (FZ)	
A	B	A	B
L943	L944	E500	E501
XXX.X	XXX.X	XXX.X	XXX.X



L943 REACTOR LEVEL A WR INCH

L944 REACTOR LEVEL B WR INCH

E500 REACTOR LEVEL A FZ INCH

E501 REACTOR LEVEL B FZ INCH

CHART

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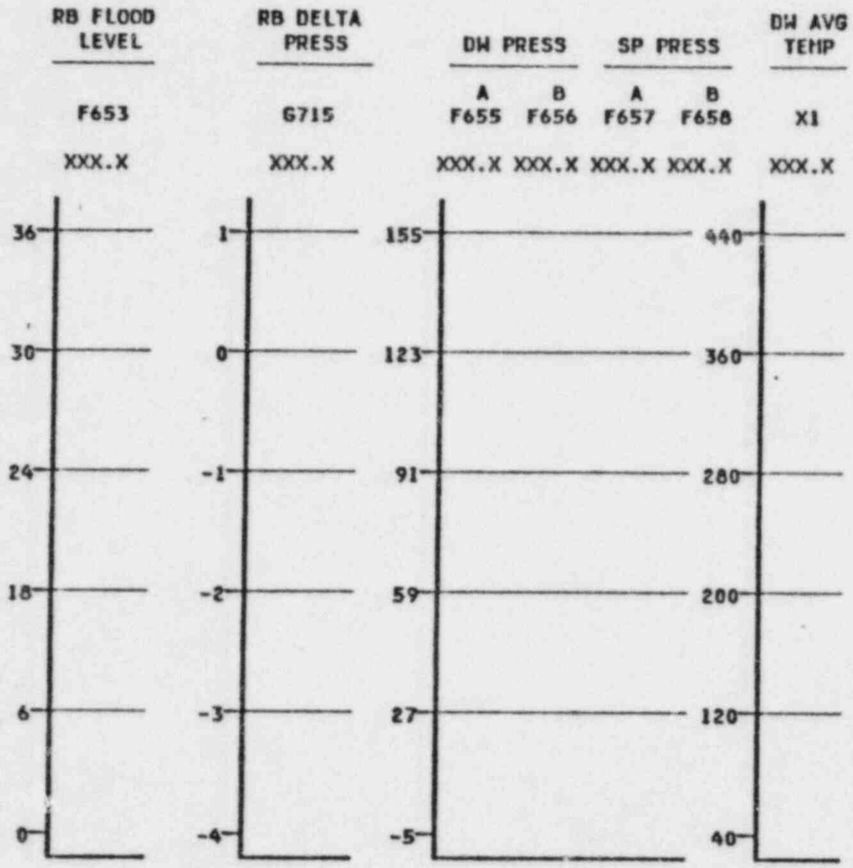
CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 18 -TIME-
HHMMSS



F653 RB FLOOD LVL A INCH

G715 REACTOR BLDG P IN H20

F655 DRYWELL WR P A PSIG

F656 DRYWELL WR P B PSIG

F657 SUPR CHMBR WR P A PSIG

F658 SUPR CHMBR WR P B PSIG

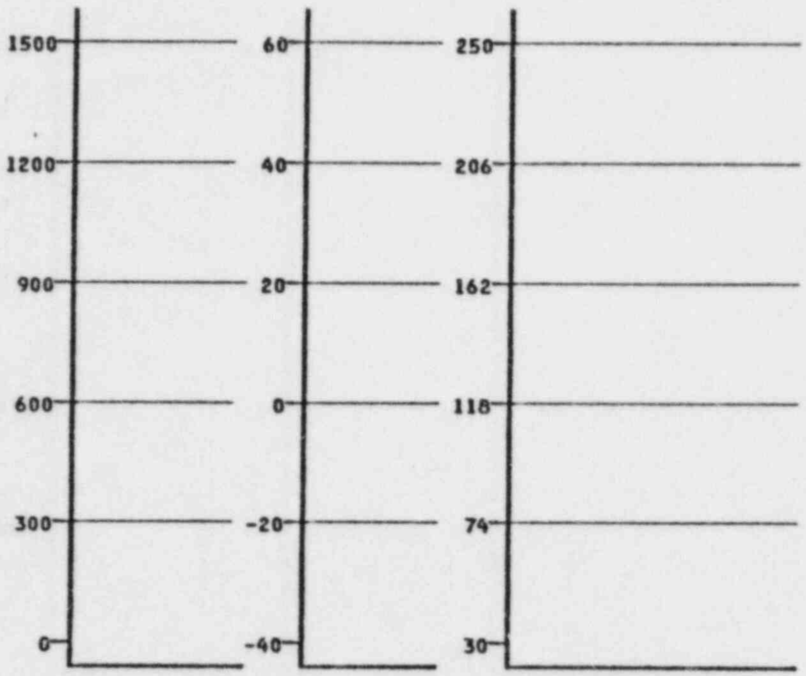
DRYWELL AVG TEMP DEGF

CHART

04/29/82

CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 19 -TIME-
HHMMSS

REACTOR PRESSURE		SP WATER LEVEL		SP WATER TEMPERATURE			
A	B	A	B	1	2	3	4
L940	L941	L927	L928	L625	L626	L925	L926
XXXX	XXXX	XXXX	YXXX	XXXX	XXXX	XXXX	XXXX



L940 REACTOR PRESSURE A PSIG

L941 REACTOR PRESSURE B PSIG

L927 SP WTR LVL A INCH

L928 SP WTR LVL B INCH

L625 SP QUAD1 2FT WTR T DEGF

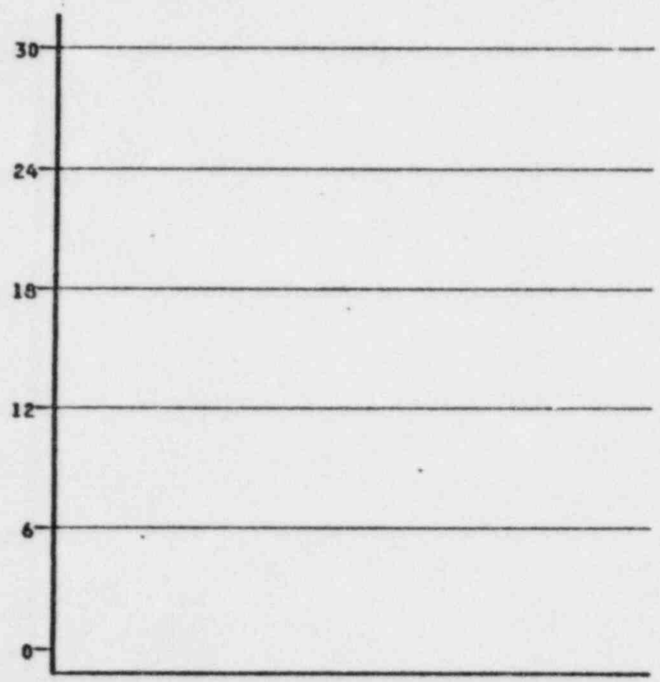
L626 SP QUAD2 2FT WTR T DEGF

L925 SP QUAD3 2FT WTR T DEGF

L926 SP QUAD4 2FT WTR T DEGF

CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 20 -TIME-
HHMMSS

DW H2 CONCENTRATION		SP H2 CONCENTRATION	
A	B	A	B
L929	L938	L939	E502
XX.XX	XX.XX	XX.XX	XX.XX



L929 DRYWELL H2 CONC A % L939 SUPR CHMBR H2 CONC A %

L938 DRYWELL H2 CONC B % E502 SUPR CHMBR H2 CONC B %

Radiation Monitoring System CRT Displays

Radioactivity Controls

Primary Containment Activity
Station Ventilation Activity
Reactor Building Standby Ventilation Activity