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August 14, 1981

Mr. Harold R. Denton
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton: ULNRC-477

DOCKET NUMBERS 50-483 AND 50-486
CALLAWAY PLANT, UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Transmitted herewith is Chapter 18 of the Callaway FSAR, the Site Addendum. This information is hereby incorporated into the Callaway Application. This will be incorporated in a future revision to the FSAR.

Very truly yours,

John K. Bryan

John K. Bryan

DJW/mdj
Enclosure

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STATE OF MISSOURI)
) S S
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John K. Bryan, of lawful age, being first duly sworn upon oath says that he is Vice President-Nuclear and an officer of Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By John K. Bryan
John K. Bryan
Vice President
Nuclear

SUBSCRIBED and sworn to before me this 14th day of August, 1981

Barbara J. Pfaff
BARBARA J. PFAFF
NOTARY PUBLIC, STATE OF MISSOURI
MY COMMISSION EXPIRES APRIL 22, 1985
ST. LOUIS COUNTY

CHAPTER 18.0

RESPONSE TO NUREG-0737
CLARIFICATION OF TMI ACTION
PLAN REQUIREMENTS

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RESPONSE TO NUREG-0737
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18.0 RESPONSE TO NUREG-0737, "CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS"

The following discussion of the SNUPPS response to NUREG-0737 is subdivided into three sections: 18.1, Operational Safety; 18.2, Siting and Design; and 18.3, Emergency Preparations and Radiation Protection. The subsections presenting the NRC guidance are verbatim quotes from NRC documents.

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18.1 OPERATIONAL SAFETY

18.1.1 SHIFT TECHNICAL ADVISOR (I.A.1.1)

18.1.1.1 NRC Guidance Per NUREG-0737

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of ensuring safe operations of the plant, including the review and evaluations of operating experience.

Clarification

The staff letter of October 30, 1979 from H. R. Denton to All Operating Nuclear Power Plants clarified the short-term STA requirements The letter indicated that the STAs must have completed

all training by January 1, 1981. This paper confirms these requirements and requests additional information.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979 letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980 letter (as revised by item I.A.1.3 of this report), an STA be available for duty on each operating shift when a plant is being operated in Modes 1-4 for a PWR and Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the October 30, 1979 letter was issued, several efforts have been made to establish, for the longer term, the minimum level of experience, education, and training for STAs. The efforts include work on the revision to ANS-3.1, work by the Institute of Nuclear

Power Operations (INPO), and internal staff efforts.

INPO recently made available a document entitled "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education, and Training." A copy of Revision 0 of this document, dated April 30, 1980, is attached as Appendix C to NUREG-0737 . Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions set forth in Sections 5 and 6 of Revision 0 to he INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter).

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their STA training program and their plans for requalification training. This description shall indicate the level of training attained by STAs by January 1, 1981 and demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter. Applicants for operating licenses shall provide the same information in this application, or amendments thereto, on a schedule consistent with the

NRC licensing review schedule.

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their long-term STA program, including qualfication, selection criteria, traininig plans, and plans, if any, for the eventual phaseout of the STA program. (Note: The description shall include a comparison of the licensee/applicant program with the above-mentioned INPO document. This request solicits industry views to assist NRC to establishing long-term improvements in the STA program. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.)

18.1.1.2 Union Electric Response

Union Electric will have a Shift Technical Advisor (STA) available on-site for each operating shift to report to the Control Room in an advisory capacity when the reactor is in modes 1-4.

The STA shall have a bachelor's degree in engineering or related science which includes or is supplemented to include sixty (60) semester hours of college level education in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, electrical and reactor control theory or a high school diploma and the foregoing 60 hours educational requirement. The STA shall also have one year of experience at a

nuclear power plant including six months onsite at the time the STA is required on shift. Nuclear power plant experience is time associated with: preoperational and startup testing activities; military, non-stationary, propulsion or production nuclear plants; reactor simulator training, or on-the-job training.

The training program for STA's will include: training in plant systems; a course in mitigating core damage; and specific training in the response and analysis of the plant for transients and accidents utilizing a SNUPPS simulator. A retraining and requalification program will be developed ninety (90) days prior to fuel load.

The requirement for the STA to be in addition to the normal shift complement is accepted as an interim NRR staff position. When the Commission's delibrations are concluded and the requirements for eliminating the STA position have been established, we fully expect to eliminate this interim commitment.

18.1.2 SHIFT SUPERVISOR'S ADMINISTRATIVE DUTIES (I.A.1.2)

18.1.2.1 NRC Guidance per NUREG-0578

Position

a. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary

management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

b. Plant procedures shall be reviewed to ensure that the duties, responsibilities, and authority of the Shift Supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room, relative to other plant management personnel. Particular emphasis shall be placed on the following:

1. The responsibility and authority of the Shift Supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The principle shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

2. The Shift Supervisor, until properly relieved,

shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

3. If the Shift Supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

c. Training programs for Shift Supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the Shift Supervisor is to provide for ensuring safety.

d. The administrative duties of the Shift Supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for ensuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

18.1.2.2 Union Electric Response

The Vice President-Nuclear, shall issue and review on an annual basis a management directive which emphasizes the responsibilities on the Shift Supervisor and clearly establishes his command duties during all operating conditions.

Plant administrative procedures shall define the duties, responsibilities and authority of Shift Supervisor, Operating Supervisors and Unit Reactors Operators. Administrative procedures shall further define the line of command for the Shift Supervisor. The Shift Supervisor reports to the Superintendent of Operations or his Assistants during normal operations and to the Emergency Duty Officer during an emergency. The Shift Supervisor is the senior licensed management representative on site during backshifts, the Shift Supervisor, is responsible to direct operation of the unit from the control room. This allows the Shift Supervisor to direct his attention to overall plant operations for which he is responsible. The Superintendent of Operations, or his Assistants, (the senior licensed management representative on the day shift) may relieve the Shift Supervisor.

In conjunction with the annual review of the management directive defining the Shift Supervisor's authorities and responsibilities, the Vice President-Nuclear shall assess the administrative duties undertaken by the Shift Supervisor. If these duties are found to detract from the Shift Supervisor's responsibility for safe operation

of the plant, they shall be delegated to other appropriate members of the plant staff.

An Operating Supervisor is present on the plant site at all times when reactor fuel is on site.

18.1.3 SHIFT MANNING (I.A.1.3)

18.1.3.1 NRC Guidance Per NUREG-0737

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 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.

Clarification

Page 3 of the July 31, 1980 letter is superseded in its entirety by the following:

Licensees of operating plans and applicants for operating licenses shall include in their administrative procedures (required by license cnditions) provisions governing required shift staffing and movement

of key individuals about the plant. These provisions are required to ensure 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 operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians, and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980, discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that ot IE Circular NO. 20-02.

In the event that overtime must be used (excluding extended periods

of shutdown for refueling, major maintenance, or major plant modifications), the following overtime restrictions should be followed:

a. An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

b. There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

c. An individual should not work more than 72 hours in any 7-day period.

d. An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation shall be authorized by the plant manager or his deputy or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at

duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

NRC encourages the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time.

The guidelines on overtime do not apply to the shift technical advisor, provided that he or she is provided sleeping accommodations and a 10-minute availability is ensured.

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 90 days after JUly 31, 1980.

See section II.A.1.2 (OF NUREG-0737) for minimum staffing and augment capabilities for emergencies."

18.1.3.2 UE Response

Union Electric shall provide the following complement on shift at all times for Modes 1-4 operation:

1 Shift Supervisor (SRO)
1 Operating Supervisor (SRO)
2 Unit Reactor Operators (RO)
2 Equipment Operators
2 Assistant Equipment Operators
1 I/C Technician
1 Rad/Chem Technician
1 Shift Technical Advisor

Additionally, Union Electric administrative procedures shall contain the following overtime restrictions for licensed personnel:

a. An individual shall not be permitted to work more than 12 hours straight (not including shift turnover time).

b. There shall be a break of at least 12 hours (which can include shift turnover time) between all work periods.

c. An individual shall not work more than 72 hours in any 7-day period.

d. An individual shall not be required to work more than

14 consecutive days without having 2 consecutive days off.

Circumstances may arise requiring deviation from the above restrictions. Any such deviations shall be permitted only in accordance with administrative procedures. All deviations shall be approved by Plant Superintendent, or his designee or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

18.1.4 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS (I.A.2.1)

18.1.4.1 NRC Guidance Per NUREG-0737

Position

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for 1 year.

Clarification

Applicants for SRO either come through the operations chain (C operator to B operator to A operator, etc.) or are degree-holding staff engineers who obtain licenses for backup purposes.

In the past, many individuals who came through the operator ranks were administered SRO examinations without first being an operator. This was clearly a poor practice and the letter of March 28, 1980 requires reactor operator experience for SRO applicants.

However, NRC does not wish to discourage staff engineers from becoming licensed SROs. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation.

In addition, in order to attract degree-holding engineers to consider the shift supervisor's job as part of their career development, NRC shold provide an alternate path to holding an operator's license for 1 year.

The track followed by a high-school graduate (a nondegreed individual) to become an SRO would be 4 years as a control room operator, at least one of which would be as a licensed operator, and participation in an SRO training program that includes 3 months on shift as an extra person.

The track followed by a degree-holding engineer would be, at a minimum, 2 years of responsible nuclear power plant experience as a staff engineer, participation in an SRO trianing program equivalent to a cold applicant training program, and 3 months on shift as an extra person in training for an SRO position.

Holding these positions ensures that individuals who will direct the licensed activities of licensed operators have had the nececssary combination of education, training, and actual operating experience prior to assuming a supervisory role at that facility.

The staff realizes that the necessary knowledge and experience can be gained in a variety of ways. Consequently, credit for equivalent experience should be given to applicants for SRO licenses.

Applicants for SRO licenses at a facility may obtian their 1-year operating experience in a licensed capacity (operator or senior operator) at another nuclear power plant. In addition, actual operating experience in a position that is equivalent to a licensed operator or senior operator at military propulsion reactors will be acceptable on a one-for-one basis. Individual applicants must document this experience in their individual applications in sufficient detail so that the staff can make a finding regarding equivalency

Applicants for SRO licenses who possess a degree in engineering or applicable sciences are deemed to meet the above requirement, provided they meet the requirements set forth in sections A.1.a and A.2 in enclosure 1 in the letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980, and have participated in a training program equivalent to that of a cold senior operator applicant.

NRC has not imposed on the 1-year experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

18.1.4.2 UE Response

UE Commits to conduct its licensed operator training and requalification programs in accordance with the requirements of 10 CFR Part 55. In addition, UE will comply with training and qualification recommendations delineated in INPO guidelines for the training of licensed personnel. The Callaway Plant Training Manual provides detailed outlines of curricula for such training sequences.

Additional discussion of UE commitments relative to training and requalification programs is presented in Section 18.1.6.

18.1.5 ADMINISTRATION OF TRAINING PROGRAMS (I.A.2.3)

18.1.5.1 NRC Guidance Per NUREG-0737

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will ensure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator

(SRO) qualifications and be enrolled in appropriate requalification programs.

Clarification

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors. The certification of instructors may, or may not, include successful completion of an SRO examination.

The purpose of the examination is to provide the NRC with reasonable assurance during the interim period that instructors are technically competent.

The requirement is directed to permanent members of training staff who teach the subjects listed above, including members of other organizations who routinely conduct training at the facility. There is no intention to require guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete an SRO examination. Nor is it intended to require a system expert, such as the instrument and control supervisor teaching the control rod drive system, to complete an SRO examination."

18.1.5.2 Union Electric Response

A structured training program for the Callaway Plant Training staff will be prepared. At least one Senior Training Supervisor or Training Supervisor will be licensed. Training Supervisors who are licensed will attend requalification lectures to maintain their proficiency.

18.1.6 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS (I.A.3.1)

18.1.6.1 NRC Guidance Per NUREG-0737

Position

Simulator examinations will be included as a part of the licensing examinations.

Clarification

The clarification does not alter the staff's position regarding simulator examinations.

The clarification does provide additional preparation time for utility companies and NRC to meet the examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated on June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site.

NRC simulator examinations normally require 2 to 3 hours. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably, these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and no later than 2 weeks after the balance of the examination.

18.1.6.2 Union Electric Response

As training programs for reactor operator qualification and requalification are developed, UE Training and Operations Department management will review the programs to ensure that they meet the requirements of 10CFR Part 55, Appendix A. Minimum overall grade for exams will be 80 percent with a minimum score of 70 percent in each category examined. Utilization of a facility simulator will be incorporated into the requalification programs for operators, as the simulator training program is developed. The Callaway Plant Training Manual also addresses in detail, the curricula for licensed personnel training and requalification.

18.1.7 EVALUATION OF ORGANIZATION AND MANAGEMENT IMPROVEMENTS OF NEAR-TERM OPERATING LICENSE APPLICANTS (I.B.1.2)

18.1.7.1 NRC Guidance Per NUREG-0694 nd NUREG-0737

Position

The licensee organization shall comply with the findings an requirements generated in an interoffice NRC review of licensee organization and management. The review will be based, in part, on an NRC document entitled "Draft Criteria for Utility Management and Technical Competence." The first draft of this document was dated February 25, 1980. The current draft was issued for interim use and public comment in September, 1980 as NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources." These draft guidelines address the organization, resources, training, and qualifications of plant staff and management (both onsite and offsite) for routine operations and the resources and activities (both onsite and offsite) for accident conditions.

The licensee shall establish a group that is independent of the plant staff but is assigned onsite to perform independent reviews of plant operational activities and a capability for evaluation of operating experiences and nuclear power plants.

Organizational changes are to be implemented on a schedule to be determined prior to fuel loading.

Corporate management of the utility-owner of a nuclear power plant shall be sufficiently involved in th perational phase activities, including plant modifications, to ensure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the opreation and maintenance of the nuclear power plant. To these ends, each utility-owner shall establish an organization, parts of which shall be located onsite, to: perform independent reviews and audits of plant activities; provide technical support to the plant staff for maintenance, modifications, operational problems, and operational analysis; and aid in the establishment of programmatic requiements for plant activities.

The licensee shall estabiish an integrated organizational arrangement to provide for the overall management of nuclear power plant operations. This organization shall provide for clear management control and effective lines of authority and communication between the organizational units involved in the management, technical support, and operation of the nuclear unit.

The key characteristics of a typical organization arrangement are:

a. Integration of all necessary functional responsibilities under a single responsible head.

b. The assignment of responsibility for the safe operation of the nuclear powr plant(s) to an upper

level executive position.

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities, including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

Clarification

The new ISEG shall not replace the plant operations review committee (PORC) and the utility's independent review and audit group as specified by current staff guidelines (Standard Review Plan, Regulatory 1.33, Standard Technical Specifications). Rather, it is an additional independent group of a minimum of five dedicated, full-time engineers, located onsite, but reporting offsite to a corporate official who holds a high-level, technically oriented position that is not in the management chain for power production. The ISEG will increase the available technical expertise located onsite and will provide continuing, systematic, and independent assessment of plant activities. Integrating the shift technical advisors (STAs) into the ISEG in some way would be desirable in that it could enhance the group's contact with and knowledge of day-to-day plant operations and provide additional expertise. However, the STA on shift is necessarily a member of the operating staff and cannot be independent of it.

It is expected that the ISEG may interface with the quality assurance (QA) organization, but preferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can, therefore, best be carried out by a group physically located onsite. However, for utilities with multiple

sites, it may be possible to perform portions of the independent safety assessment function in a centralized location for all the utilities' plants. In such cases, an onsite group still is required, but is may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirement for establishing an ISEG is being applied only to applicants for operating licenses in accordance with Action Plan Item I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to ascertain whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Action Plan Item I.B.1.1)."

18.1.7.2 Union Electric Response

Union Electric has established an organization whose authorities and responsibilities are consistent with the guidance in NUREG-0731. The UE organization provides for integration of all functional responsibilities under a single responsible head and the responsibility for safe operation of the nuclear plant is assigned to an upper level executive position. The separation of key organizations such as Quality Assurance and Radiation Protection from operating pressures is provided. Members of the organization exceed the minimum educational requirements set forth in NUREG 0731 and referenced in Regulatory Guide 1.8 and ANS 3.1. Minimum requirements

of nuclear power experience are also satisfied through various training programs, on-the-job experience, and other pertinent experience considered on a case-by-case basis, within the fuel load time frame.

Union Electric has committed to the establishment of Independent Safety Engineering Group. The ISEG will be headed by the Superintendent ISEG reporting offsite to the Manager of Nuclear Engineering. The ISEG Superintendent will direct a group of graduate engineers. It is expected that this group will consist of four engineers in addition to the superintendent. This group will have expertise in nuclear, mechanical, electrical, and chemical engineering as well as some experience in nuclear operations. This group will perform independent reviews of plant operational activities and will evaluate operating experiences at nuclear plants. The ISEG Charter has been submitted to the NRC as part of the responses to the Management Audit items of July 14, 1981.

18.1.8 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS (I.C.1)

18.1.8.1 NRC Guidance Per NUREG-0737

Position

In letter of September 13 and 27, October 10 and 30, and NOvember 9, 1979, the Office of Nuclear Reactor Regulation required licensees of

operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedures guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (also refer to Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of tansients and accidents were to be completed in early 1980, and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established, however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of the review of these matters on Babcock and Wilcocx (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (refer to Table C.1, items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, items 4, 12, 17, 18, 19, 20; and Table C.3, items 6, 35, 37, 38, 41, 47, 55, 57).

Clarification

The letter of September 13, 27, October 10 and 30, and November 8, 1979 required that procedures and operator training be developed for transients and accidents. The initiating events to be considered should include the events presented in the Final Safety Analysis Report (FSAR): loss of instrumentation buses and natural phenomena

such as earthquakes, floods, and tornadoes. The purpose of this paper is to clarify the requirements and add additional requirements for the reanalysis of transients and accidents and inadequate core cooling.

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed. Specifically, the staff has found a lack of justification for the approach used (i.e., symptom-event-, of function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for the operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the

failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure events include:

a. Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator.

b. Failure of main and auxiliary feedwater.

c. Failure of high-pressure reactor coolant makeup system.

d. An anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety/relief valve, or main feedwater.

e. Operator errors of omission or commission.

The analyses should be carried out far enough into the event to ensure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long-term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate

core cooling procedure, including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

a. A detailed description of the methodology used to develop the guidelines.

b. Associated control function diagrams, sequence-of-event diagrams, or others, if used.

c. The bases for multiple and consequential failure considerations.

d. Supporting analysis, including a description of any computer codes used.

e. A description of the applicability of any generic results to plant specific applications.

Owners' Group or vendor submittals may be referenced as appropriate to support this reanalysis. If Owners' Group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines, the

staff will continue the pilot monitoring of emergency procedures described in Task Action Plan Item I.C.8 (NUREG-0660). For PWRs, this will involve review of the loss-of-coolant, steam-generator tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequacy of each PWR vendor's guidelines will be identified to each NTOL during the emergency-procedure review. Since the analysis and guidelines submitted by the General Electric Company (GE) Owners' Group that comply with the requirements stated above have been reviewed and approved for trial implementation on six plants with applications for operating licenses pending, the interim program for BWRs will consist of trial implementation of these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human-factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control-room design, the time required for component and system response, clarity of procedural actions, and control-room personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in Task Action Plan Item I.C.9, will be made available. At that time, the reviews currently being conducted on NTOLs under Item I.C.8 will be discontinued, and the review required for applicants for operating licenses will be as described in the long-term plan. Depending on the information

submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule, provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under Item I.C.8 to revise its emergency procedures again prior to the final implementation date for Item I.C.9. The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is the use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issued under 10 CFR 50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations

from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small-break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9."

18.1.8.2 Union Electric Response

Through participation in the Westinghouse Owners' Group (WOG), the SNUPPS utilities have been involved in the development of Westinghouse guidelines for accidents that exceed existing design basis and guidelines for inadequate core cooling. Guidelines are to be submitted to the NRC for review and approval, and after NRC approval is obtained, the guidelines are used for the preparation of generic and/or plant-specific emergency operating and inadequate core cooling procedures.

The WOG has supported development of additional guidelines for comparison to existing guidelines for emergency operation. Events to be reconsidered in the light of NUREG-0737 guidance (i.e., multiple failures) are:

- Large LOCA
- Small LOCA
- Feedline break
- Steamline break
- Steam generator tube rupture

The WOG had pla[illegible] to finalize the expanded guidelines, which include th[illegible] for detailed procedures to mitigate inadequate core cooling by late summer, 1981. The WOG has submitted an update of Westinghouse Topical Report WCAP-9691, which used even tree methodology to extend a review of analyzed accidents to include certain multiple failure considerations. WCAP-9691 ws updated to expand the Westinghouse Reference Operating Instruction set through considerations of extended coverage provided by current Emergency Operating Instruction Guidelines. A significant number of the original WCAP-9691 event sequences were provided with additional "procedural coverage" as a result of the evaluation commissioned by the Owners' Group. The most recent correspondence between the Chairman of the WOG and the NRC indicates that the NRC has been apprised of the WOG's overall approach and has met with representatives of the WOG in order to discuss expanded emergency operating and inadequate core cooling guidelines. However, the NRC has not yet approved the guidelines.

Union Electric will develop emergency operating procedures consistent with the WOG guidelines for the events enumerated above. Union Electric will evaluate each guideline and develop plant emergency operating procedures specifically applicable to Callaway Plant.

18.1.9 SHIFT RELIEF AND TURNOVER PROCEDURES (I.C.2)

18.1.9.1 NRC Guidance Per NUREG-0579

Position

The licensee shall review and revise, as necessary, the plant procedure for shift and relief turnover to ensure the following:

1. A checklist shall be provided for the oncoming and offgoing control-room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console. What to check and criteria for acceptable status shall be included in the checklist.

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement. (This shall be

recorded as a separate entry on the checklist.)

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by itself could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A sytem shall be established to evaluate the effectiveness of the shift and relief turnover procedures (for example, periodic independent verification of system alignments)."

18.1.9.2 UE Response

Callaway Plant administrative procedures shall define specific shift relief and turnover procedures for licensed operators. Turnover checklists shall be developed to include the following information:

a. Means to assure that critical plant parameters are within allowable limits.

b. Means to assure the availability and proper alignment

of all safety-related systems.

c. Means to identify any activities impacting Technical Specifications.

d. A clear record of transfer of the command function on each shift.

Additional checklists or logs shall be provided for equipment operators to record any safety-related equipment in a degraded mode or that in a state of operation which could initiate an operational transient involving safety-related equipment.

The adequacy of shift relief and turnover procedures shall be evaluated periodically as directed by administrative procedures.

18.1.10 SHIFT SUPERVISOR'S RESPONSIBILITIES (I.C.3)

This item is discussed in Section 18.1.2, Shift Supervisor Administrative Duties.

18.1.11 CONTROL ROOM ACCESS (I.C.4)

Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (E.G., operations supervisor, shift supervisor,

and control room operators), to technical advisors who may be requested or required to support the operation, and the predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct comand of operations, including those who report to stations outside the control room."

18.1.11.2 Union Electric Response

Union Electric shall develop an administrative procedure which includes limitations on access to the control room. In addition to the access control provisions available via the plant security systems, other restrictions shall be imposed by administrative procedures. During normal operations, access shall be limited to

those individuals whose preserce is necessary to carry out assigned functions. In an emergency situation, access to the control room shall be limited by the shift supervisor to the operating shift complement, Plant Superintendent, Assistant Plant Superintendent, Superintendent of Operations, Assistant Superintendent of Operations, one NRC representative, and additional management of support personnel deemed necessary to effectively handle the situation.

Union Electric shall provide administrative procedures which define the line-of-command in the control room. The Shift Supervisor is in overall command of the plant and the Operating Supervisor is in direct control room command. Union Electric shall provide administrative procedures which define lines of communication and authority for Callaway Plant management "who report to stations both within and outside the control room."

18.1.12 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

18.1.12.1 NRC Guidance Per NUREG-0737

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to ensure that operating information pertinent to plant safety originating, both

within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

1. Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

2. Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures, operating orders);

3. Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, and health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

4. Provide means to ensure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through

routine training and retraining programs;

5. Ensure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

6. Provide suitable checks to ensure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and

7. Provide periodic internal audit to ensure that the feedback prog [illegible] unctions effectively at all levels.

Clarification

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to ensure that important information on operating experience originating both within and outside the origanization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures, training, and retraining program.

Those involved in the assessment of operating experience will review

information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE bulletins, circulars, notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be dealth with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to opreating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to ensure that operations management personnel would be directly involved in the process. In many other cases, however, important information will be come available which should be brought to the attention of operators and other personnel for their general information to ensure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealth with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information."

18.1.12.2 Union Electric Response

UE has committed to provide an Independent Safety Engineering Group whose responsibilities are delineated in Section 18.1.7. One of the tasks with which ISEG is chartered is independent review of reactor

operating experience at both the Callaway Plant and at other facilities.

In addition, UE has revised an administrative procedure entitled "Review of Recent Reactor Operating Experience." The purpose of this procedure is to address the review of IE Bulletins, circulars, Notices, "Nuclear Notepad", "See-In" and Nuclear Power Experience Reports for feedback and dissemination of appropriate operating experiences affecting plant operation and training. This procedure will be submitted for NRC review.

Until the ISEG is established the Engineering Department at Callaway shall be responsible for dissemination of information as per the procedure AP-A-20. A draft of this procedure was submitted to the NRC as part of the response to Management Audit items of July 14, 1981.

18.1.13 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES (I.C.6)

18.1.13.1 NRC Guidance Per NUREG-0737

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to ensure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the

quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring, if required, will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with Item I.D.3.

Clarification

Item I.C.6 of the U.S. Nuclear Regulatory Commission Task Action Plan (NUREG-0660) and Recommendation 5 of NUREG-0585 propose requiring that licensees' procedures to be reviewed and revised, as necessary, to ensure that an effective system of verifying the correct performance of operating activities is provided. An acceptable program for verification of operating activities is described below.

The American Nuclear Society had prepared a draft revision to ANSI Standard N18.7-1972 (ANSI 3.2), "Administrative Controls and Quality Assurance For the Operational Phase of Nuclear Power Plants." A

second proposed revision to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," which is to be issued for public comment in the near future, will endorse the latest draft revision to ANSI 3.2 subject to the following supplemental provisions.

1) Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.

2) In lieu of any designated senior reactor operator (SRO), the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of equipment control measures, such as the tagging of equipment.

4) Equipment control procedures shoulkd include assurance that control room operators are informed of changes in equipment status and the effects of such changes.

5) For the reutrn-to-service of equipment important to safety, a second qualified operator should verify

proper system alighment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

Note: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The Staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring, as discussed in Task Action Plan Item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will re reduced. However, the need for such verification will not be eliminated in all instances.

18.1.13.2 Union Electric Response

UE is committed to having procedures which ensure an effective system of verifying correct preformance of Callaway's operating activities. Such procedures shall be reviewed for applicability to this section of NUREC-0737 (I.C.6). Procedures addressing the return to service of Safety-related equipment will require two authorized personnel initials verifying system alignment unless functional testing can be performed without compromising plant safety.

Administrative procedures will address the transfer of operating information from the off-going to the on-going shift personnel to ensure that status of equipment is understood. These administrative procedures will be complete 90 days before fuel load.

Training for these procedures will be complete 90 days before fuel load.

A rough draft procedure has been submitted to meet the requirements of Item I.C.6 as described in NUREG-0737.

18.1.14 NSS VENDOR REVIEW OF PROCEDURES (I.C.7)

18.1.14.1 NRC Guidance Per NUREG-0660

"Applicants for near-term operating licenses will be required to obtain NSSS vendor review of their low-power and power-ascension test, and emergency procedures as a further verification of the adequacy of the procedures."

18.1.14.2 Union Electric Response

Union Electric commits to a review by Westinghouse of pertinent specific low-power, ascension, and emergency procedures to provide further verification of their adequacy.

18.1.15 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR

NEAR-TERM OPERATING LICENSE APPLICANTS (I.C.8)

18.1.15.1 NRC Guidance Per NUREG-0737

Position

The NRC will conduct an interdisciplinary and interoffice audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steamline break, or steam-generator tube rupture).

The licensee should correct, before full-power operation, any deficiencies in the emergency procedures, as necessary, based on the NRC audit.

18.1.15.2 Union Electric Response

Union Electric will transmit emergency operating procedures to the Nuclear Regulatory Commission at their request. After review and comment by the NRC, Union Electric will evaluate suggested changes in the procedures and incorporate those deemed necessary to to assure proper operation during emergency conditions.

18.1.18 SPECIAL LOW POWER TESTING AND TRAINING (I.G.1)

18.1.18.1 NRC Guidance Per NUREG-0737

NUREG-0694, "TMI-Related Requirements for New Operating Licenses," requires applicants for a new operating license to define and commit to a special low-power testing program approved by the NRC staff, to be conducted at power levels no greater than 5 percent, for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training. This requirement must be met before fuel loading.

Position

The staff position was stated in a letter to the applicants dated November 14, 1980. This letter stated that the program should provide for the following:

> "Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should experience the initiation, maintenance, and recovery from natural circulation mode, using nuclear heat to simulate decay heat. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.
>
> These tests should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to

establish and maintain natural circulation with or without onsite and offsite power, and the ability to uniformly borate and cool down to hot shutdown conditions, using natural circulation. The latter demonstration may be performed using decay heat following power ascension and vendor acceptance tests, and need only to perform at those plants for which the tests has not been demonstrated at a comparable prototype plant."

18.1.18.2 Union Electric Response

Natural circulation testing will be conducted at Callaway to insure the following areas are satisfied.

1. Training - Each cold-licensed RO (RO or SRO who perform RO or SRO duties respectively) will participate or be simulator-trained in the initiation, maintenance and recovery from natural circulation mode. Operators will be able to recognize when natural circulation has stabilized and will be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits. These tests will be conducted in so far as possible to include all available licensed operators. All licensed operators will be trained in these same areas on the Callaway simulator.

2. Testing - The tests will demonstrate the following plant characteristics: Length of time required to stablize natural circulation, core flow distribution, ability to establish and maintain natural circulation. The simulator will have full capability of simulating natural circulation, using W data initially. When these tests are accomplished on the plant, actual data will be inserted into the program.

3. Procedure Validation - These tests will make maximum practical use of Callaway written plant procedures to validate the completeness and accuracy of the procedures.

4. If natural circulation tests have been performed at comparable plants, the tests will be repeated at Callaway only insofar as necessary to insure the requirements of the training area are complete.

18.2.4 TRAINING FOR MITIGATING CORE DAMAGE (II.B.4)

18.2.4.1 NRC Guidance Per NUREG-0737

Position

The staff reuires that the applicants develop a program to ensure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged." The training program shall include the following topics:

a. Incore Instrumentation

   1. Use of fixed or movable incore detectors to determine the extent of core damage and geometry changes.

   2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

b. Excore Nuclear Instrumentation (NIS)

   Use of NIS for determination of void information; void location basis for NIS response as a function of core temperatures and density changes.

c. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual versus indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.

   a) Determination of pressurizer level if all level transmitters fail.

   b) Determination of letdown flow with a clogged filter (low flow).

   c) Determination of other reactor coolant system parameters if the primary method of measurement has failed.

d. Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leaktight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primar water; time to failure.

e. Radiation Monitoring

1. Response of process and area monitors to severe damage; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (over ranged detector); expected accuracy of detectors at different locations; use of detectors to determine the extent of core damage.

2. Methods of determining dose rate inside the containment from measurements taken outside the containment.

f. Gas Generation

1. Methods of H generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

2. H flammability and explosive limit, sources of O in continment or reactor coolant system.

18.2.4.2 Union Electric Response

Union Electric will have presented a course on mitigating core damage to licensed operators 90 days before fuel load. The topics covered will be the topics outlined in section 18.2.4.1.

18.2.15 REQUESTS BY NRC INSPECTION AND ENFORCEMENT BULLETINS (II.K.1)

18.2.15.1 NRC Guidance Per NUREG-0694

Position

"(C.1.5)-Review all valve positions, positioning requirements, positive controls, and related tests and maintenance procedures to assure proper ESF functioning. See Bulletins 79-06A Item 8, 79-06B Item 7, and 79008 Item 6 in Reference 11 (NUREG-0560).

"(C.1.10)-Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to ensure that operability status is known. See Bulletins 79-05A Item 10, 79-06A Item 10, 79-06B Item 9, and 79-08 Item 8 in Reference 11 (NUREG-0560).

"(C.1.17)-For Westinghouse-designed reactors, trip the pressurizer low-level coincident signal bistables, so that safety injection would be initiated when the pressurizer low-pressure setpoint is reached

regardless of the pressurizer level. See Bulletine 79-06A and Revision 1, Item 3 in Reference 11 (NUREG-0560).

18.2.15.2 Union Electric Response

The development and review of procedures for testing, maintenance, and system operation for the Callaway facility are being carried out as a joint effort between Union Electric, Kansas Gas and Electric, the SNUPPS project organization, and other consultants.

The item related to the safety injection logic is not applicable to the Callaway design (see Figure 7.2-1, Sheet 8).

18.3 EMERGENCY PREPARATIONS AND RADIATION PROTECTION

18.3.1 UPGRADE EMERGENCY PREPAREDNESS (III.A.1.1)

18.3.1.1 NRC Guidance Per NUREG-0694

Position

"Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA (Federal Emergency Management Agency) findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations. This requirement shall be met before issuance of a full-power license.'

18.3.1.2 Union Electric Response

The Callaway Plant Radiological Emergency Response Plan (RERP) was submitted as Appendix 13.3A in Revision 3 to the Callaway Plant FSAR Site Addendum. NRC questions on this submittal have been received (R. L. Tedesco letters dated 6/29/81, 7/15/81 and 7/22/81).

18.3.2 UPGRADE EMERGENCY SUPPORT FACILITIES (III.A.1.2)

18.3.2.1 NRC Guidance Per NUREG-0578 and NUREG-0694

(A) ONSITE TECHNICAL SUPPORT CENTER (NUREG-0578, Item 2.2.2.b)

Position

"Each operating nuclear power plant shall maintain an onsite technical support center (TSC) separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support to reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-build conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC."

Clarification (NRC Letter dated November 9, 1979)

1. By January 1, 1980, each licensee should meet items 1-7 that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

a. Establish a TSC and provide a complete description.

b. Provide plans and procedures for engineering/management support and staffing of the TSC.

c. Install dedicated communications between the TSC and the control room, near-site emergency operations center, and the NRC.

d. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets or evacuation to the control room).

e. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters necessary for assessment in the TSC.

f. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical informatin during an emergency. The center should be located on site, i.e., within the plant security boundary. The greater the distance from the control room, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size and Staffing

The TSC should be large enough to house 25 persons, necessary engineering data, and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4. Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The shift technical advisor should be consulted on the "Notification of Unusual Event." However, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirments but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the compability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to ensure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To ensure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be given to ensure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) shold be available to permit the assessment of:

a. Plant Safety System Parameters for:

1) Reactor Coolant System

2) Secondary System (PWRs)

3) ECCS Systems

4) Feedwater and Makeup Systems

5) Containment

b. In-Plant Radiological Parameters fo

1) Reactor Coolant System

2) Containment

3) Effluent Treatment

4) Release Paths

c. Offsite Radiological

1) Meteorology

2) Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display. Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the emergency operations center, the NRC, the reactor vendor, etc.

9. Structural Integrity

a. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

b. Since the center need not be designed to the same stringent requirements as the control room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the Technical Support Center personnel from radiological hazards, including direct radiation and airborne contaminants, as per General Design Criterion 19 and SRP 6.4.

a. Licensee should ensure that personnel inside the Technical Support Center (TSC) will not receive doses in excess of those specified in GDC-19 and SRP 6.4 (i.e., 5 rem whole-body and 30 rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

b. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

c. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be

followed, except that the systems do not have to be redundant, seismic, instrumented in the control room, or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52, and the HEPAs do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

d. Dose reduction measures such as breathing apparatus and potassium iodide tablets cannot be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available."

(B) ONSITE OPERATIONAL SUPPORT CENTER (NUREG-0578, Item 2.2.2.c)

Position

"An area to be designated as the onsite Operational Support Center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to

reflect the existence of the center and to extablish the methods and lines of communication and management."

(C) NEAR-SITE EMERGENCY OPERATION FACILITY (NUREG-0694)

Position

"Designate a near-site Emergency Operations Facility (EOF) with communications with the plant to provide evaluation of radiation releases and coordination of all onsite and offsite activities during an accident.

Provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite Technical Support Center, and direct display of radiological and meteorological parameters."

18.3.2.2 Union Electric Response

The emergency response facilities for the Callaway Plant were designed using guidance of NUREG-0696, Final Report, entitled "Functional Criteria for Emergency Response Facilities." The facilities will be completed prior to fuel load.

Technical Support Center

The Technical Support Center (TSC) at the Callaway Site is located within the protected area and immediately adjacent to the on-site

building that contains the offices of the Plant Superintendent and the plant support staff. This building is termed the Service Building at the Callaway Site. The location of this building and the Technical Support Center is shown in Figures 18.3.2-1.

This location for the TSC was selected because there is no suitable space within the power block and because:

- This location facilitates activation of the TSC, since the persons designated to man the TSC have their offices in the Service Building.

- There is ready accessibility to plant data available in the Service Building which is not stored in the TSC (e.g., vendor manuals).

During normal plant operations the TSC will be readily available to the onsite engineering staff, which is quartered in the Service Building.

The distance from the TSC to the control room is approximately 700 feet and respectively, for Callaway. The walking time is estimated to be about three minutes.

The TSC, shown in Figure 18.3.2-2 is a one story building of 5,000 square feet located at grad level. The walls are reinforced concrete 10 inches thick and the roof is reinforced concrete 6 inches thick. The structural design is in conformance with the Uniform

Building Code. Within the TSC are 2900 square feet of working space that contain displays of plant status, meeting and discussion areas, communications facilities, and document storage. The remaining 2100 square feet within the TSC are occupied by a mechanical equipment room, which contains HVAC equipment, a dedicated diesel-generator for the TSC, a computer room, and limited toilet, kitchen and access facilities. This is sufficient space for at least 25 persons, including five NRC personnel. For any extended durationof TSC operation, additional toilet, locker room, and kitchen facilities in the Service Building will be available.

The HVAC system for the TSC supplies outside air appropriately cooled or heated and has provisions to isolate inlet air and to operate in a filtered recirculation mode if radiation levels are high. The filter train contains HEPA and charcoal filters. Switchcover to the filtered recirculation mode is manual.

Radiation monitoring in the TSC consists of one area radiation monitor in the main work area, with a range of 1 to 10 mr/hr, and a radioiodine monitor which will detect concentrations as low as 10 c/cc. The radioiodine monitor will read the concentration in the inlet air to the TSC in whichever mode the HVAC system, open or recirculating, is employed.

Electric power to the TSC in a post-accident situation is normally provided by a transformer from off-site power. In addition, there is a dedicated standby diesel generator, rated at 218 kva, that is started manually utilizing dedicated battery power. The diesel

generator has sufficient capacity to power all TSC loads, including the computer system, the communications system, HVAC and lighting. Essential equipment in the TSC is also provided with power supplies to keep the equipment operable during a power interruption, as for example, loss of offsite power after activation of the TSC and until the standby diesel generator is started and assumes load. The computer system and communications systems have uninterruptable power supplies (UPS). Emergency lighting consisting of self-contained battery units is also provided in the TSC.

Protective clothing, breathing apparatus, and personnel radiation monitors to permit up to 10 persons to function within radiation areas will be accessible to the TSC.

The conditions for manning the TSC are described in general terms in the Emergency Plans for the Callaway and Wolf Creek sites. These are contained in Appendix 13.3 A of the Callaway FSAR Site Addendum. Detailed procedures are in the process of being developed, as Emergency Plan implementing procedures.

Operations Support Center

The location designated for the post-accident Operations Support Center (OSC) at Callaway is the Service Building. This location is shown in Figures 18.3.2-2 and 18.3.2-3. This location provides ample space for assembly of personnel and has communications to the Control Room and TSC. Maintenance equipment, tools and protective clothing are also available. There is no special radiation protection because

the OSC would not be utilized at times when outsie radiation levels are high.

Emergency Operations Facility

At Callaway the Emergency Operations Facility (EOF) is located approximately 1 mile from the plant, as shown in Figure 18.3.2-4. It is a one-story building of _3,000 square feet, which is divided roughly equally into (1) a Recovery Center and supporting services and (2) an Emergency Control Center and supporting services.

The EOF working space is sufficient for at least 35 persons, consisting of 25 persons designated by the licensee including state and local officials, 9 persons from the NRC and one person from FEMA. The structural design of the EOF is in conformance to the Uniform Building Code. Walls are concrete, approximately 10 inches thick and the roof consists of double-T pre-cast concrete sections with a minimum concrete thickness of approximately 6 inches. The structure provides radiation shielding equivalent to a protection factor greater than 5.

The HVAC system for the EOF is similar to that of the TSC, except it contains only HEPA and no charcoal filters.

Radiation monitoring in the EOF is the same as described for the TSC.

Electric power for the EOF is normally provided by a transformer from offsite power. In addition there is a dedicated standby diesel-

generator to operate the EOF in the event of loss of offsite power. The standby diesel generator is started manually, utilizing dedicated standby power. As in the TSC, the computer and communications systems are each provided with a Uninterruptible Power Supply. Emergency lighting consisting of self-contained battery units is also provided in the EOF.

Because of the radiation protection factors provided by the building structures, it is the judgement of Union Electric that a backup EOF, as described by NUREG-0696, is not necessary.

Emergency Response Facilities Information System (ERFIS)

The Emergency Response Facilities Information System (ERFIS) consists of the:

1. The plant data acquisition system which supplies data to the:

    a. Control room CRT's

    b. Safety Parameter Display System (SPDS) in the control room

    c. Control room printer

    d. ERFIS communications processor

2. ERFIS communications processor which supplies data to the:

   a. TSC subsystem

   b. Nuclear Data Link (if required)

3. TSC subsystem which supplies data to the:

   a. TSC peripheral equipment

   b. SPDS in the TSC

   c. EOF peripheral equipment

   d. SPDS in the EOF

A. <u>Safety Parameter Display System (SPDS)</u>

The Safety Parameter Display System is being designed jointly by a group of Westinghouse NSSS utilities of which SNUPPS is a member.

The control room SPDS is being designed for a 99% availability, and will not be seismically qualified. A separate concentrated seismically qualified backup SPDS in the control room is not provided.

The requirement to install separate additional seismic displays is in conflict with the design criteria of Reg. Guide 1.97 which encourages that the operator use normal operating displays during accidents. This use of existing displays assures that the operator will always get information to perform critical and normal operating functions from the same location. The SPDS, concentrates a minimum set of plant parameters to aid the operator in the rapid detection of abnormal operating events. However, it is reasonable to use the normal qualified displays as a backup for this purpose.

B. Nuclear Data Link

The Nuclear Data Link has no definition at this time since the protocol and definition required has not been established by the NRC. Hardware provisions have been made to implement this on the Communication processor when details are available.

C. Technical Support Center System (TSCS)

The TSCS conceptually consists of a CPU, CRT's and a high speed printer. SPDS displays along with the displays currently available in the control room will be abailable in the TSC. The complete data base from the plant data acquisition system along with the data from the Radiation Release Information System and the Post Accident Sampling System will be available. This data base includes the Regulatory Guide 1.97 parameters.

The TSCS is being designed for a 99% availability. Details of instrumentation quality, accuracy, and reliability have not yet been established.

D. Emergency Operations Facility System (EOFS)

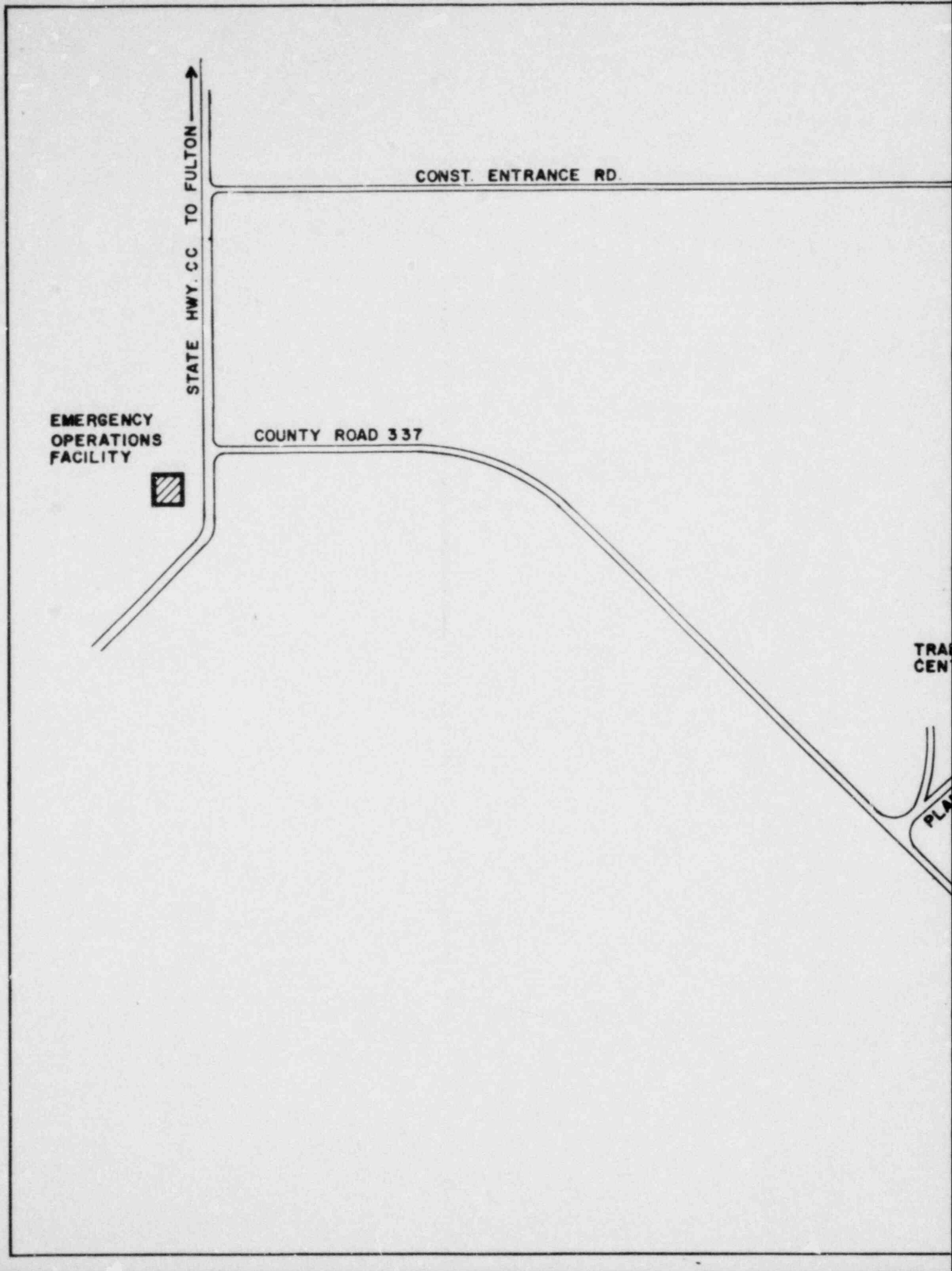
The EOFS conceptually consists of CRT's and low speed printers. SPDS displays along with the displays currently available in the control room will be available in the TSC. Current plans are to have the same data base at the TSC available at the EOF. The TSC CPU drives the EOFS peripherals through a data link. As a result of the distance to the EOF, the data link has to be compatible with the environment.

The EOFS is being designed for a 99% availability. Details of instrumentation quality, accuracy, and reliability have not yet been established.

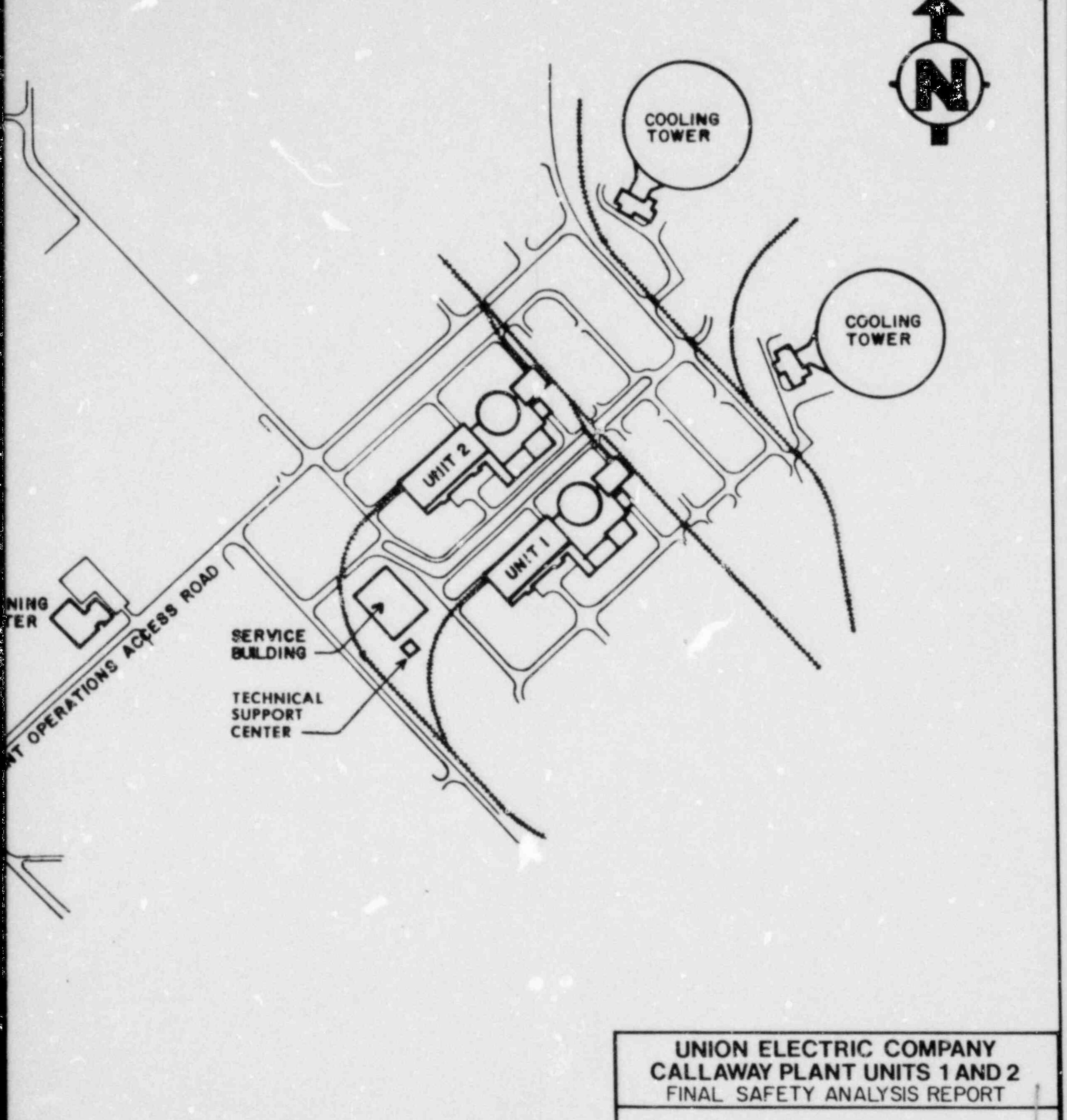
Task Functions for the TSC and EOF

Union Electric has submitted The Callaway Plant Radiological Emergency Response Plan to the NRC as part Appendix 13.3A of the FSAR Site Addenda. Details of task functions and manning can be found in this plan.

18.3.3 IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM (III.A.2)



STATE HWY. CC TO FULTON
CONST. ENTRANCE RD.
EMERGENCY OPERATIONS FACILITY
COUNTY ROAD 337



UNION ELECTRIC COMPANY
CALLAWAY PLANT UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FIGURE 18.3.2-1
EMERGENCY RESPONSE FACILITIES
LOCATIONS

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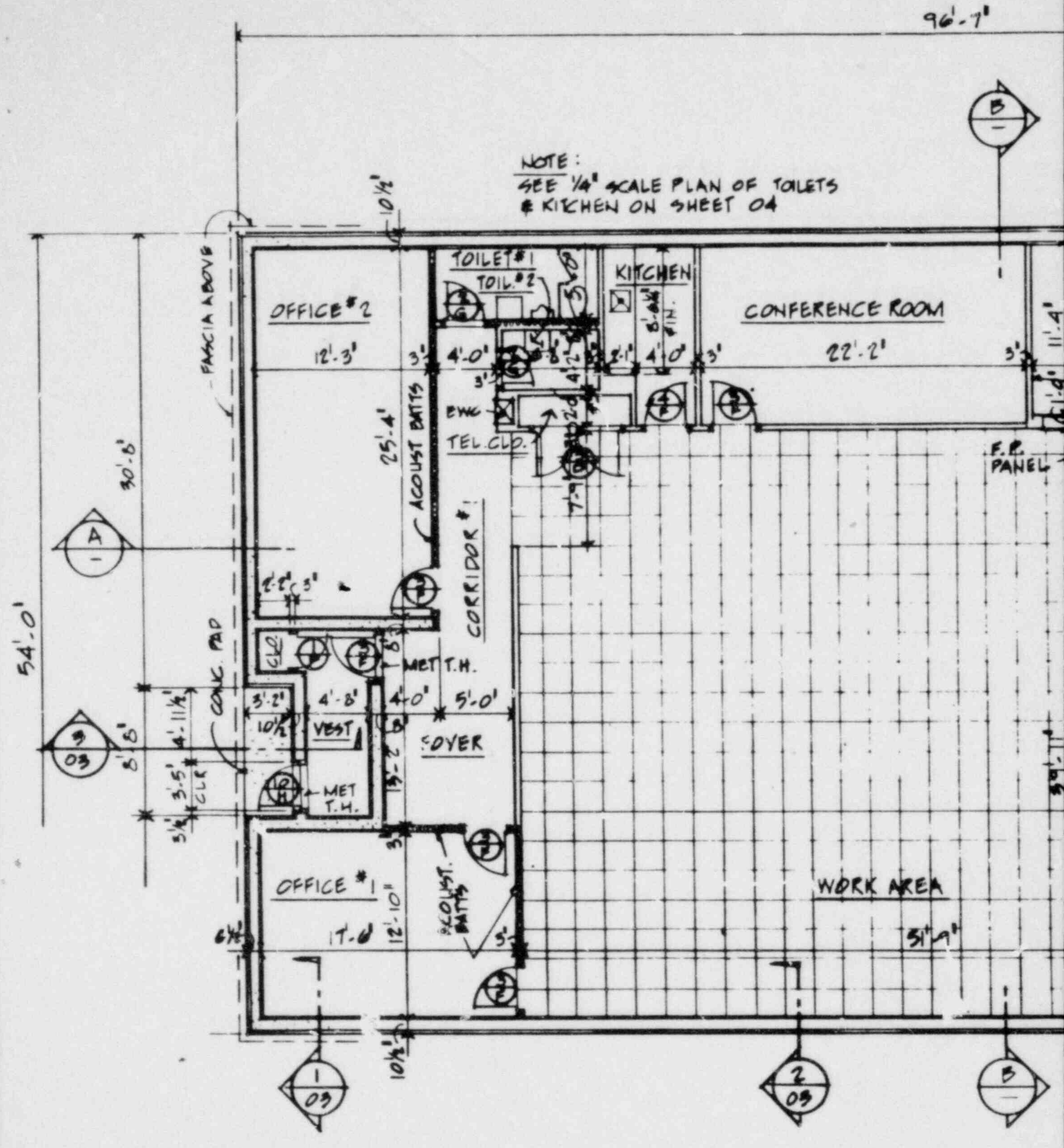

FLOOR PLAN

1/8" = 1'-0"

FIN. FLOOR EL. = 840.0'

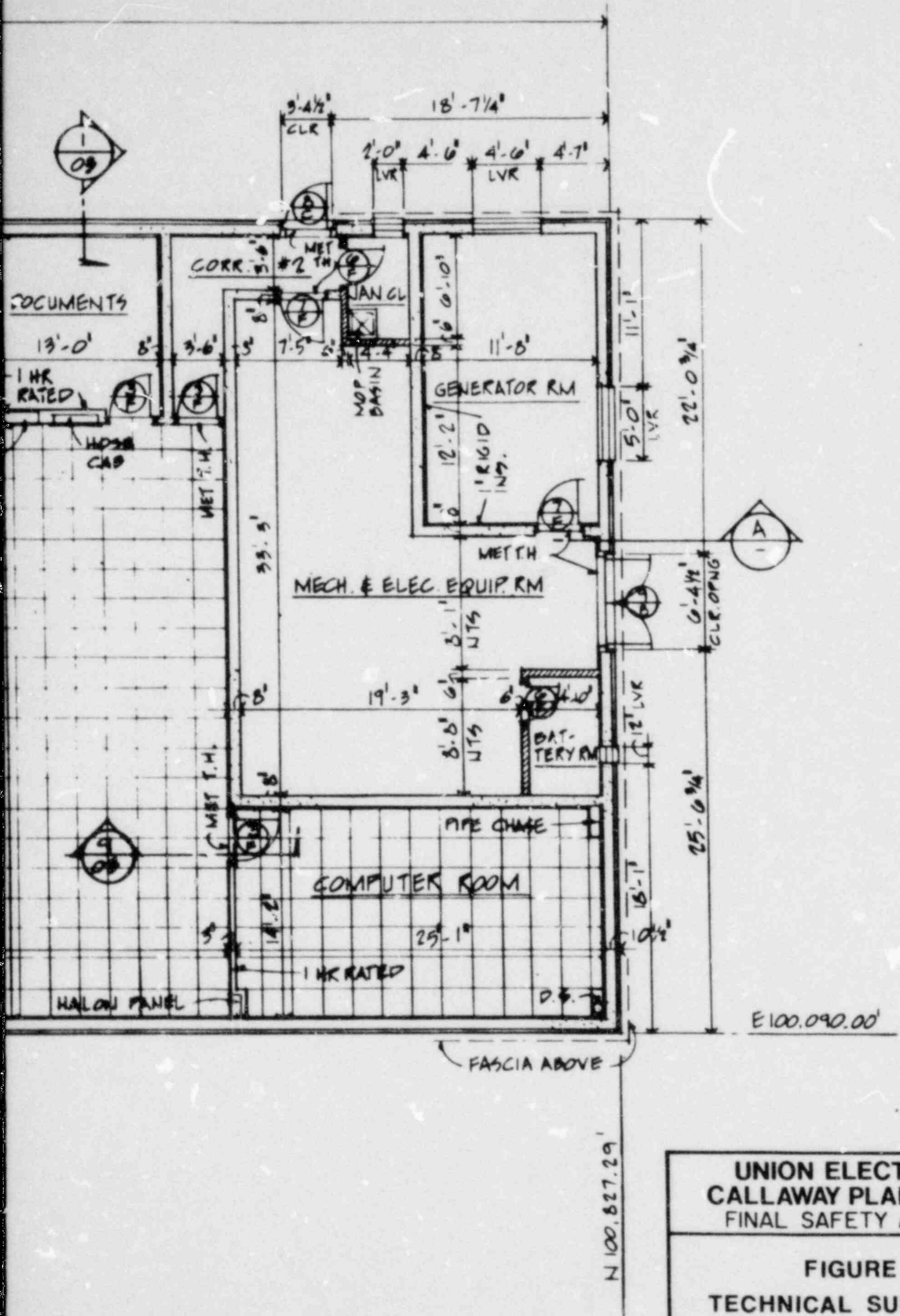

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FIGURE 18.3.2-2
TECHNICAL SUPPORT CENTER LAYOUT

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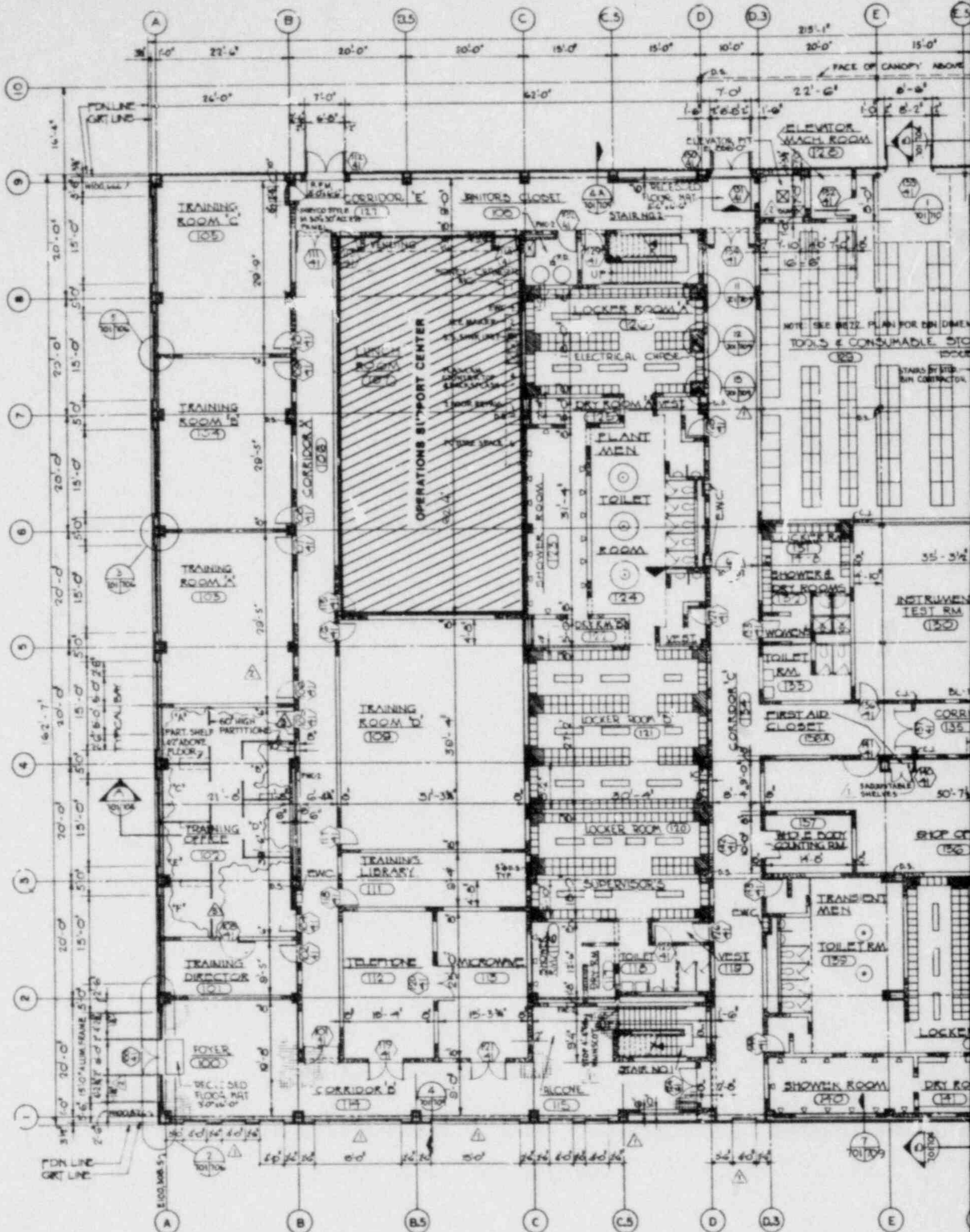
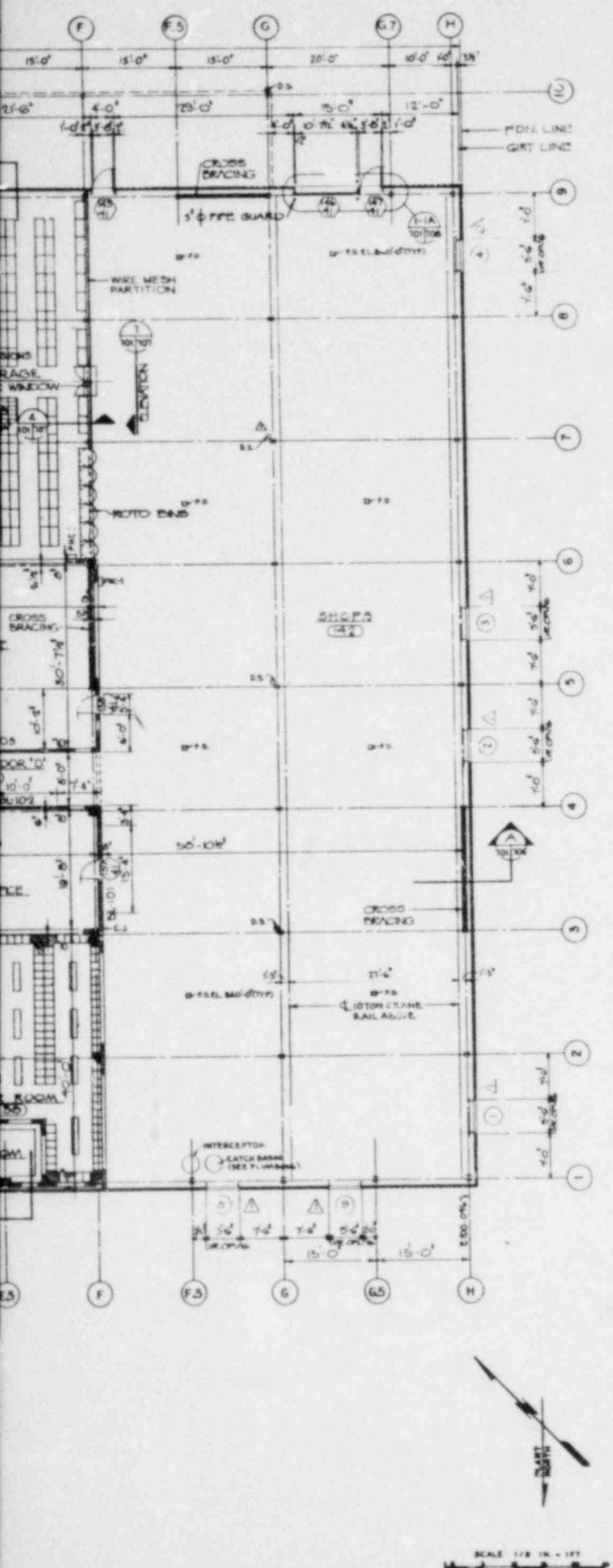


FIRST FLOOR PLAN
SC. 1/8"=1'-0"
FINISH FLOOR EL. 830'-0" UNLESS OTHERWISE NOTED



ABBREVIATIONS			
BMS.	BEAMS	LVR.	LOUVER
BLK.	BLOCK	L.P.	LOW POINT
CAB.	CABINET	MAS.	MASONRY
CLG.	CEILING	MAX.	MAXIMUM
CEM.	CEMENT	MECH.	MECHANICAL
CONC.	CONCRETE	MET.	METAL
CONT.	CONTINUOUS	MIN.	MINIMUM
CORR.	CORRIDOR	NO.	NUMBER
DET.	DETAIL	O.C.	ON CENTER
DIM.	DIMENSION	OPNG.	OPENING
DN.	DOWN	PART.	PARTITION
D.S.	DOWNSPOUT	PL.	PLATE
DWG.	DRAWING	REF.	REFERENCE
EL.	ELEVATION	RM.	ROOM
EWC.	ELECTRIC WATER COOLER	R.H.	ROUND HEAD
EQUIP.	EQUIPMENT	SHT.	SHEET
EXH.	EXHAUST	SPECS.	SPECIFICATIONS
EXIST.	EXISTING	STL.	STEEL
FIN.	FINISH	STR.	STRUCTURAL
FL.	FLOOR	TPTN.	TOILET PARTITION
FTG.	FOOTING	T.O.S.	TOP OF STEEL
FDN.	FOUNDATION	TYP.	TYPICAL
GA.	GAUGE	VERT.	VERTICAL
GALV.	GALVANIZED	VEST.	VESTIBULE
H.P.	HIGH POINT	W/	WITH
HORIZ.	HORIZONTAL	S.STL.	STAINLESS STEEL
I.D.	INSIDE DIAMETER	R.O.	ROOF OPENING ⚠
INSUL.	INSULATION		
LAM. PLAS.	LAMINATED PLASTIC		

SYMBOLS

ALL SYMBOLS INDICATED ARE NOT NECESSARILY USED. ALL SYMBOLS ARE FOR SECTIONS & DETAILS ONLY UNLESS OTHERWISE NOTED.

- CONCRETE
- EARTH
- INSULATION, RIGID
- METAL, ROLLED SHAPES
- WOOD, ROUGH
- DOOR NUMBER / DESIGN PACKAGE
- SECTION OR DET. DESIGNATION / SHT. NO. WHERE SECTION OR DET. IS DRAWN. / SHT. NO. WHERE SECTION OR DET. IS REFERENCED.

GENERAL NOTES:

1. ROOM 110 NOT USED
2. SEE SHEET 111 FOR 1/4" SCALE PLANS OF TOILET ROOMS
3. SEE SHEET 110 FOR 1/4" SCALE PLAN & SECTION STAIR NO. 1 & NO. 2.
4. DOOR 123 NOT USED
5. SEE REFLECTED CEILING PLANS FOR GEN. PARTITION HEIGHTS & FIRE RATINGS

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FIGURE 18.3.2-3

OPERATIONS SUPPORT CENTER

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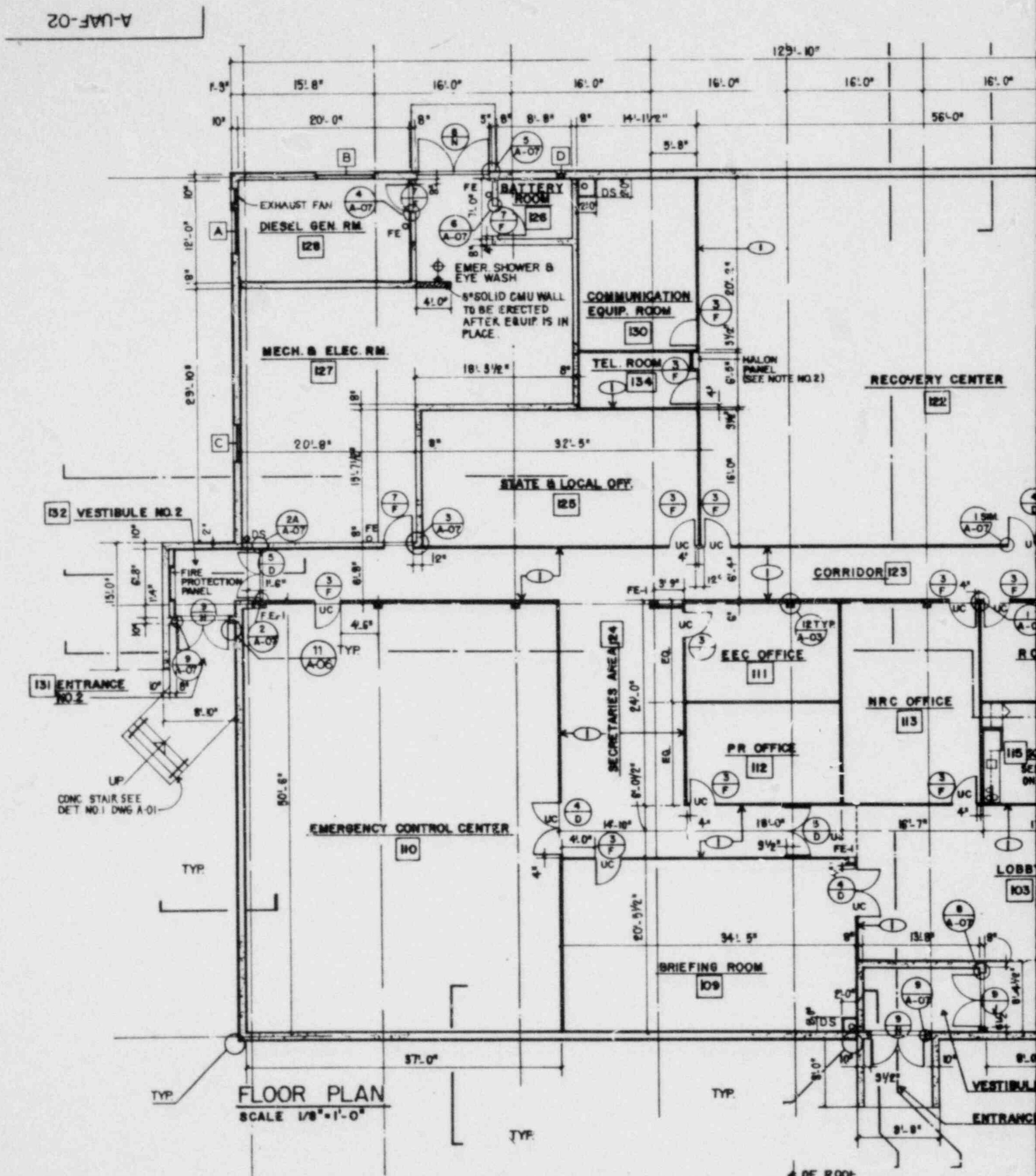

FLOOR PLAN
SCALE 1/8" = 1'-0"

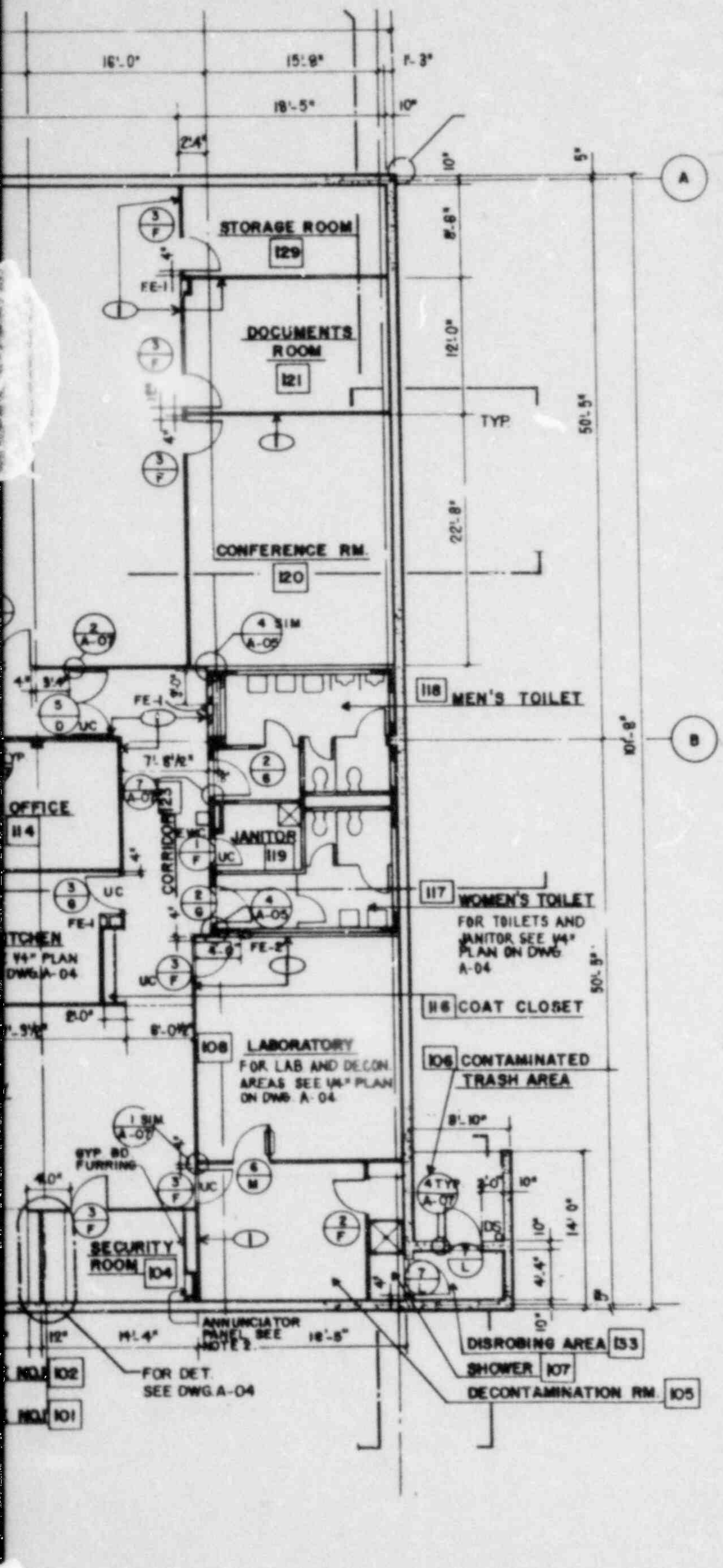

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FIGURE 18.3.2-4
EMERGENCY OPERATIONS FACILITY LAYOUT

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18.3.3.1 NRC Guidance Per NUREG 0737

"Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radilogical emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Classification

In accordance with Task Action Plan item III.A.1.2, "Upgrade Emergency Preparedness," each nuclear power facility was required to immediately upgrade its emergency plans with criteria provided October 10, 1979, as revised by NUREG-0654 (FEMA-REP-1, issued for interim use and comment, January 1980). New plans were submitted by January 1, 1980, using the October 10, 1979 criteria. Reviews were started on the upgraded plans using NUREG-0654. Concomitant to these actions, amendments were developed to 10 CFR Part 50 and Appendix E to 10 CFR Part 50, to provide the long-term implementation requirements. These new rules were issued in the Federal Register on August 19, 1980, with an effective date of November 3, 1980. The revised rules delineate requirements for emergency preparedness at nuclear reactor facilities.

NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluationof Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides detailed items to be included in the

upgraded emerency plans and, along with the revised rulse, provides for meteorological criteria, means for providing for a prompt notification to the population, and the need for emergency response facilities see Item III.A.1.2 (of NUREG-0737).

Implementation of the new rules levied the requirement for the licensee to provide procedures implementing the upgraded emergency plants to the NRC for review. Publication of Revision 1 to NUREG-0654 (FEMA-REP-1) which incorporates the many public comments received is expected in October 1980. This is the document that will be used by NRC and FEMA in their evaluation of emergency plans submitted in accordance with the new NRC rules.

NUREG-0654, Revision 1; NUREG-0696, "Functional Criteria for Emergency Response Facilities," and the amendments to 10 CFR Part 50 and Appendix E to 10 CFR Part 50 regarding emergency preparedness, provide more detailed criteria for emergency plans, design, and functional criteria for submission of upgraded emergency plans for installation of prompt notification systems. These revised criteria and rules supersede previous Commission guidance for the upgrading of emergency preparedness at nuclear power facilities.

Revision 1 to NUREG-0654, "Critria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides meteorological criteria to fulfill, in part, the standard that "Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use" (see 10 CFR

650.47). The posi''on in Appendix 2 to NUREG-0654 outlines four essential elements that can be categorized into three functions: measurements, assessment, and communications.

Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Measurements Programs in Suppo:t of Nuclear Power Plants," has been adopted to provide guidance criteria for the primary meteorological measurements program consisting of a primary system and secondary system(s) where necessary, and a backup system. Data collected from these systems are intended foruse in the assessment of the offsite consequences of a rdiological emergency condition.

Appendix 2 to NUREG-0654 delineates two classes of assessment capabilities to provide input for the evaluation of offsite consequences of a radiological emergency condition. Both classes of capabilities provide input to decisions regarding emergency actions. The Class A capability should provide information to determine the necessity for notification, sheltering, evacuation, and, during the initial phase of a radiological emergency, making confirmatory radiological measurements. The Class B capability should provide information regarding the placement of supplemental meteorological monitoring equipment, and the need to make additional confirmatory radiological measurements. The Class B capability shall identify the areas of contaminated property and food stuff requiring protective measures and may also provide information to determine the necessity for sheltering evacuation.

Proposed Revision 1 to Regulatory Guide 1.23 outlines the set of meteorological measurements that should be accessible from a system that can be interrogated; the meteorological data should be presented in the prescribed format. The results of the assessments should be accesible from this system; this information should incorporate human-factors engineering in its display to convey the essential information to the initial decision makers and subsequent management team. An integrated and radiological monitoring information with the environmental transport to provide direct dose consequence assessments.

Requirements of the new emergency-preparedness rules under Paragraphs 50.47 and 50.54 and the revised Appendix E to Part 50 taken together with NUREG-0654 Revision 1 and NUREG-0696, when approved for issuance, go beyond the previous requirements for meteorological programs. To provide a realistic time frame for implementation, a staged schedule has been established with compensating actions provided for interim measures."

18.3.3.2 Union Electric Response

See response to 18.3.1.2.

18.3.5 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

18.3.5.1 NRC Guidance per NUREG-0737

Position

a. Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

b. Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Clarification

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

a. The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

For applicants with fuel-loading dates prior to January 1, 1981, provide by fuel loading (until January 1, 1981) the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer

(SCA). The SCA window should be calibrated to the 365 KeV of iodine-131, using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

18.3.5.2 Union Electric Response

The emergency plans for Callaway and the SNUPPS report on the emergency response facilities will address this subject.