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CHAPTER 18.0

RESPONSE TO NUREG-0737

CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

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CHAPTER 18.0

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

The following discussion of the Union Electric 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.

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 Manager. 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 managers 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 the 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 qualification, selection criteria, training 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 in 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 <u>Union Electric Response</u>

General

The NRC issued a Policy Statement on Engineering Expertise on Shift in October 1985. The Policy Statement permits either of two options to be used to implement long term goals toward upgrading the qualifications and training of operating staffs. These are Option 1: Combined SRO/STA Position and Option 2: Continued use of STA Position. Either of these options may be used to meet the requirements of NUREG-0737, Item I.A.1.1. Also, either Option 1 or 2 may be used on each shift. The complete requirements for the options are as follows:

Option 1: Combined SRO/STA Position

This option is satisfied by assigning an individual with the following qualifications to each operating shift crew as one of the SROs (preferably the Shift Manager) required by 10 CFR 50.54 (m)(2)(i):

1. Licensed as a senior operator on the nuclear power unit(s) to which assigned and;

2. Meets the STA training criteria of NUREG-0737, Item I.A.1.1, and one of the following educational alternatives:

    a. Bachelor's degree in engineering from an accredited institution;

    b. Professional Engineer's license obtained by the successful completion of the PE examination;

    c. Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences; or

    d. Bachelor's degree in a physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

Option 2: Continued Use of STA Position

This option is satisfied by placing on each shift a dedicated Shift Technical Advisor (STA) who meets the STA criteria of NUREG-0737, Item I.A.1.1. The STA should assume an active role in shift activities. For example, the STA should review plant logs, participate in shift turnover activities, and maintain an awareness of plant configuration and status.

As stated in NUREG-0737 and the background to the NRC Policy Statement (discussed below), the requirement for an STA qualified person in the power plant in addition to an SRO licensed Shift Manager was intended to be a temporary requirement until the qualifications of the Shift Manager and senior operators are upgraded and control boards are reviewed and modified to make information and controls more useful to the operators. This is consistent with the industry consensus established by INPO standard GPG-01, "Nuclear Power Plant Shift Technical Advisor Position Description Qualifications, Education and Training" which refers to the fact of this position being "eliminated" when certain additional actions are completed. The December 17, 1983 approved copy of ANSI/ANS 3.1 which also referred to this position as "interim".

The NRC staff has completed the review of the Union Electric qualification program developed to address NUREG-0737, Item I.A.1.1. The program and NRC staff review results are discussed below.

Man-Machine Interface Upgrade

The man-machine interface in the control room has been upgraded by means of an extensive control room design review which included human factors input. For a detailed description of this effort refer to Section 18.1.16 and 18.1.17.

Operator qualification upgrading in accordance with NUREG-0737, Item I.A.1.1 is discussed in Section 13.2.

Union Electric has either a senior operator or an engineer, who meets the STA training criteria of NUREG-0737, for each operating shift to report to the control room when the reactor is in Modes 1-4.

The individuals fulfilling the STA requirement 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.

The individuals fulfilling the STA requirement shall also have one year of experience at a nuclear power plant including six months onsite at the time the individual 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 STA qualification program includes: 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 the Callaway Plant simulator. A retraining and requalification program has been developed.

The Director, Nuclear Operations is responsible for the qualification of the STAs.

The senior operators report to the Manager, Nuclear Operations. The description and functions of the senior operators are further detailed in Sections 13.1.2.2 and 13.1.3.1.

18.1.1.3 Conclusion

Union Electric's actions to upgrade qualification of operating staff and the man-machine interface for control room personnel are consistent with the intent and specifics of the long-term resolution of NUREG-0737, Item I.A.1.1.

18.1.2 SHIFT MANAGER 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 Manager 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 Manager 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 manager 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 Manager 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 Manager 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 Manager, 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 Manager shall be specified.

3. If the Shift Manager 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 Manager shall emphasize and reinforce the responsibility for safe operation and the management function the Shift Manager is to provide for ensuring safety.

D. The administrative duties of the Shift Manager 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 Senior Vice President and Chief Nuclear Officer, issues and reviews on an annual basis a management directive which emphasizes the responsibilities on the Shift Manager and clearly establishes his command duties during all operating conditions.

Plant administrative procedures define the duties, responsibilities and authority of Shift Manager, Operating Supervisors and Unit Reactors Operators. Administrative procedures further define the line of command for the Shift Manager. The Shift Manager reports to the Director, Nuclear Operations or the Manager, Nuclear Operations during normal operations and to the Emergency Duty Officer during an emergency. The Shift Manager is the senior licensed management representative on site during weekends and backshifts. The Shift Manager is responsible to direct operation of the unit. This allows the Shift Manager to direct his attention to overall plant operations for which he is responsible. The Director, Nuclear Operations, or the Manager, Nuclear Operations, shall designate senior reactor operators who may relieve the Shift Manager.

In conjunction with the annual review of the management directive defining the Shift Manager's authorities and responsibilities, the Senior Vice President and Chief Nuclear Officer shall assess the administrative duties undertaken by the Shift Manager. If these duties are found to detract from the Shift Manager's responsibility for safe operation of the plant, they shall be delegated to other appropriate members of the plant staff.

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

18.1.2.3 Conclusion

Union Electric's commitment to the establishment and annual review of management directives defining the responsibilities and authority of the Shift Manager and to the implementation of the training programs in accordance with 10 CFR 55 meets the intent of NUREG-0737, Item I.A.1.2.

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 plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to 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 prevent situations where fatigue could reduce the ability of operating personnel to keep the reactor in a safe condition. The controls established should assure that, to the extent practicable, personnel are not assigned to shift duties while in a fatigued condition that could significantly reduce their mental alertness or their decision making ability. The controls shall apply to the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel).

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

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

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

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

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

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

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.

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.

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

18.1.3.2 UE Response

Shift staffing is discussed in Section 13.1 of the FSAR. Union Electric has an administrative procedure governing shift manning and movement of key individuals. Unexpected absences are also addressed in plant procedures. Additional information on staffing requirements is contained in Technical Specifications, Table 16.12-1 and the RERP.

18.1.3.3 Conclusion

In 2009, new regulation 10 CFR 26, Subpart I, became effective. This regulation addresses worker fatigue and specifies limits and controls on working hours. The requirements of 10 CFR 26, Subpart I, supersede prior worker fatique guidance. It distinguishes between work hour controls and fatigue management and strengthens the requirements for both. Union Electric (dba AmerenUE) observes the work hour restrictions required by 10 CFR Part 26, Subpart I. In addition, Union Electric continues to meet the intent of NRC’s guidance for minimum shift complement.

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 manager's job as part of their career development, NRC should 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 training 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 necessary 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 obtain 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 has committed to conduct its licensed operator training and requalification programs in accordance with the requirements of 10 CFR Part 55 and INPO accredited programs as referenced in Section 13.2. In addition, UE complies with the training and qualification recommendations delineated in INPO guidelines for the training of licensed personnel. The Callaway Plant Training Procedures provide 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.

18.1.4.3 Conclusion

Union Electric has committed to comply with 10 CFR 55 requirements for operator licensing and requalification programs and INPO accredited programs.

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

All operator license program instructors are required to participate in requalification programs, which include simulator training, for licensed operators. In addition, licensed training instructors engaged in operator training participate in the following activities:

a. periodic onshift assignments

b. review of facility operating and emergency operating procedures as they are developed

c. participation in instructor certification programs, such as those proposed by INPO

This material is discussed further in Section 13.2.2.

18.1.5.3 Conclusion

Union Electric has committed to comply with NRC guidance relative to the training and qualification of nuclear training staff, and meets the legal requirements of 10 CFR 55 where SRO licenses are required.

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

Utilization of the Callaway simulator is incorporated into the requalification programs for operators as discussed in Section 13.2. The Callaway Plant Training procedures also address in detail, the curricula for licensed personnel training and requalification.

18.1.6.3 Conclusion

Union Electric has committed to NUREG-0737 guidance with regard to the NRC's revised scope and criteria for license examinations. In addition, UE has incorporated simulator examinations into it's training program. These commitments comply fully with NRC guidance as stated in NUREG-0737 relative to licensing examinations.

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 and NUREG-0737 Position

The licensee organization shall comply with the findings and 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 the operational phase activities, including plant modifications, to ensure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the operation 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 requirements for plant activities.

The licensee shall establish 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 power 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 it 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

The Director, Nuclear Oversight is responsible for independent reviews of Callaway Plant operational activities. The Performance Improvement Department is responsible for the Callaway Plant Operating Experience Program.

18.1.7.3 Conclusions

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 Nuclear Oversight 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 and on-the-job experience.

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 transients 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 Wilcox (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 letters 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, or 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."

The NRC issued additional requirements and guidance relative to this issue in Supplement 1 to NUREG-0737 (Generic Letter 82-33). The requirements and guidance in Supplement 1 to NUREG-0737 replaced the corresponding requirements in NUREG-0737.

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 have been submitted to the NRC for review and have been approved. These guidelines have been used for the preparation of Callaway specific Emergency Operating Procedures 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 has completed the expanded guidelines which include the bases for detailed procedures to mitigate inadequate core cooling. The WOG has submitted an update of Westinghouse Topical Report WCAP-9691, which used event tree methodology to extend a review of analyzed accidents to include certain multiple failure considerations. WCAP-9691 was updated to expand the Westinghouse Reference Operating Instruction set through consideration 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. By letter dated December 24, 1984, the NRC staff has approved the WOG guidelines (Westinghouse Emergency Response Guidelines, Revision 1) for implementation.

Union Electric has developed emergency operating procedures consistent with the WOG guidelines for the events enumerated above. UE has evaluated each guideline and developed plant emergency operating procedures specifically applicable to Callaway Plant. Consistent with the requirements of Supplement 1 to NUREG-0737, UE has submitted a Procedures Generation Package (PGP) for NRC review. The NRC review of the PGP resulted in the submittal of additional information (ULNRC-659, 9/7/83) regarding verification and validation of the EOPs. Information to address the process that was used to derive the instrumentation and control characteristics was submitted to NRC via ULNRC-1242, 1/14/86 and ULNRC-1323, 7/9/86.

18.1.8.3 Conclusions

Union Electric's commitment to develop emergency operating procedures based on the WOG guidelines is consistent with NUREG-0737, Supplement 1 requirements. UE has tailored the Westinghouse guidelines so that they are directly applicable to Callaway Plant. In addition, UE has provided emergency operating procedures to Westinghouse for review (See Section 18.1.14) in accordance with NUREG-0737 guidance.

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 manager 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 system 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 define specific shift relief and turnover procedures for licensed operators. Turnover checklists 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 are provided for operations technicians (or assistant operations technicians) 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.

18.1.9.3 Conclusions

Union Electric has satisfied NRC guidance relative to shift relief/turnover procedures through commitments contained in administrative procedures.

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

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

18.1.11 CONTROL ROOM ACCESS (I.C.4)

18.1.11.1 NRC Guidance Per NUREG-0578

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 manager, 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 command of operations, including those who report to stations outside the control room."

18.1.11.2 Union Electric Response

Union Electric has developed 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 are imposed by administrative procedures. During normal operations, access is limited to those individuals whose presence is necessary to carry out assigned functions. In an emergency situation, access to the control room shall be limited by the Shift Manager to the operating shift complement, EDO, Plant Director; Director, Nuclear Operations; Managers, Operations; one NRC representative, and additional management and support personnel deemed necessary to effectively handle the situation.

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

18.1.11.3 Conclusions

Union Electric has established a procedure which clearly defines the line of authority in the control room during normal and emergency situations. In addition, UE has established a procedure that clearly defines restrictions on control room access during normal and emergency conditions. These procedures comply with the NRC guidance specified in NUREG-0578.

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 program functions 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 organization 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 dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to operating 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 become 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 dealt 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 <u>Union Electric Response</u>

The Superintendent, Performance Improvement is responsible for the review, evaluation and dissemination of operating experience.

18.1.12.3 Conclusion

Union Electric's internal programs and procedure for the review, evaluation, and dissemination of operating experience gained at Callaway Plant and at other operating facilities fulfill the NRC's NUREG-0737 guidance relative to feedback and evaluation of operating experience.

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 manager 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 should include assurance that control room operators are informed of changes in equipment status and the effects of such changes.

5. For the return-to-service of equipment important to safety, a second qualified operator should verify proper system alignment 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 be reduced. However, the need for such verification will not be eliminated in all instances.

18.1.13.2 Union Electric Response

UE has developed procedures which ensure an effective system of verifying correct performance of Callaway's operating activities. These procedures were reviewed for applicability to this section of NUREG-0737 (I.C.6). Procedures addressing the return to service of safety-related equipment require two authorized personnel initials verifying system alignment unless functional testing can be performed without compromising plant safety.

Administrative procedures address the transfer of operating information from the off-going to the on-going shift personnel to ensure that status of equipment is understood. In addition, UE complies with the recommendations of Regulatory Guide 1.33 as discussed in the OQAM.

18.1.13.3 Conclusion

Union Electric's administrative controls for performance of operating activities satisfy the guidance of NUREG-0737, Item I.C.6.

18.1.14 NSSS 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

Specific low-power, ascension, and emergency procedures were reviewed by Westinghouse to provide further verification of their adequacy.

18.1.14.3 Conclusion

Union Electric's actions with regards to the Westinghouse review of selected procedures comply with the guidance of NUREG-0737.

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

NRC has conducted an indepth review of the WOG guidelines (as discussed in Section 18.1.8.1). Since UE agreed to use the WOG guidelines, this item is considered closed.

18.1.16 CONTROL ROOM DESIGN REVIEW (I.D.1)

18.1.16.1 NRC Guidance Per NUREG-0737

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to the issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Clarification

NRR is presently developing human engineering guidelines to assist each licensee and applicant in performing detailed control room review. A draft of the guidelines has been published for public comment as NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation." The due date for comments on this draft document was September 29, 1980. NRR will issue the final version of the guidelines as NUREG-0700, by February 1981, after receiving, reviewing, and incorporating substantive public comments from operating reactor licensees, applicants for operating licenses, human factors engineering experts, and other interested parties. NRR will issue evaluation criteria, by July 1981, which will be used to judge the acceptability of the detailed reviews performed and the design modifications implemented.

Applicants for operating licenses who will be unable to complete the detailed control room design review prior to the issuance of a license are required to perform a preliminary control room design assessment to identify significant human factors problems. Applicants will find it of value to refer to draft document NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation," in performing the preliminary assessment. NRR will evaluate the applicants' preliminary assessments, including the performance by NRR of onsite review/audit. The NRR onsite review/audit will be on a schedule consistent with licensing needs and will emphasize the following aspects of the control room:

1. The adequacy of information presented to the operator to reflect plant status for normal operation, anticipated operational occurrences, and accident conditions.

2. The groupings of displays and the layout of panels.

3. Improvements in the safety monitoring and human factors enhancement of controls and control displays.

4. The communications from the control room to points outside the control room, such as the onsite technical support center, remote shutdown panel, and offsite telephone lines, and to other areas within the plant, for normal and emergency operation.

5. The use of direct rather than derived signals for the presentation of process and safety information to the operator.

6. The operability of the plant from the control room with multiple failures of nonsafety-grade and nonseismic systems.

7. The adequacy of operating procedures and operator training with respect to the limitations of instrumentation displays in the control room.

8. The categorization of alarms, with unique definition of safety alarms.

9. The physical location of the Shift Supervisor's office, either adjacent to or within the control room complex.

Prior to the onsite review/audit, NRR will require a copy of the applicant's preliminary assessment and additional information, which will be used in formulating the details of the onsite review/audit.

18.1.16.2 Union Electric Response

Refer to the following documentation:

SLNRC 81-51, dated June 26, 1981
SLNRC 82-016, dated March 16, 1982
SLNRC 83-063, dated November 30, 1983
SLNRC 84-019, dated February 2, 1984
SLNRC 84-048, dated March 21, 1984
ULNRC - 822, dated May 15, 1984
ULNRC - 855, dated June 29, 1984
SLNRC 84-121, dated October 10, 1984
SLNRC 84-134, dated December 21, 1984
SLNRC 85-11, dated April 1, 1985
SLNRC 85-12, dated April 26, 1985
SLNRC 85-16, dated May 24, 1985
Callaway SSER #3 I.D.1, p. 22-3, 4, 5
Callaway SSER #4 I.D.1, p. 22-1, 2, 3, 4, 5, 6, 7

The Callaway Control Room Design Review was accepted by the NRC by letter dated August 27, 1985 - B.J. Youngblood to D.F. Schnell.

18.1.16.3 Conclusions

Based on the DCRDR activities and the NRC staff review results as detailed above, Union Electric acceptably complies with the regulatory requirements for control room design review.

18.1.17 PLANT SAFETY PARAMETER DISPLAY SYSTEM (I.D.2)

18.1.17.1 NRC Guidance Per NUREG-0696

The purpose of the safety parameter display system (SPDS) is to assist control room personnel in evaluating the safety status of the plant. The SPDS is to provide a continuous indication of plant parameters or derived variables representative of the safety status of the plant. The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions. The functional criteria for the SPDS presented in this section are applicable for use only in the control room.

It is recognized that, upon the detection of an abnormal plant status, it may be desirable to provide additional information to analyze and diagnose the cause of the abnormality, execute corrective actions, and monitor plant response as secondary SPDS functions.

As an operator aid, the SPDS serves to concentrate a minimum set of plant parameters from which the plant safety status can be assessed. The grouping of parameters is based on the function of enhancing the operator's capability to assess plant status in a timely manner without surveying the entire control room. However, the assessment based on SPDS is likely to be followed by confirmatory surveys of many non-SPDS control room indicators.

Human factors engineering shall be incorporated in the various aspects of the SPDS design to enhance the functional effectiveness of control room personnel. The design of the primary or principal display format shall be as simple as possible, consistent with the required function, and shall include pattern and coding techniques to assist the operator's memory recall for the detection and recognition of unsafe operating conditions. The human-factored concentration of these signals shall aid the operator in functionally comparing signals in the assessment of safety status.

All data for display shall be validated where practicable on a realtime basis as part of the display to control room personnel. For example, redundant sensor data may be compared, the range of a parameter may be compared to predetermined limits, or other quantitive methods may be used to compare values. When an unsuccessful validation of data occurs, the SPDS shall contain means of identifying the impacted parameter(s). Operating procedures and operator training in the use of the SPDS shall contain information and provide guidance for the resolution of unsuccessful data validation. The

objective is to ensure that the SPDS presents the most current and accurate status of the plant possible and is not compromised by unidentified faulty processing or failed sensors.

The SPDS shall be in operation during normal and abnormal operating conditions. The SPDS shall be capable of displaying pertinent information during steady-state and transient conditions. The SPDS shall be capable of presenting the magnitudes and the trends of parameters or derived variables as necessary to allow rapid assessment of the current plant status by control room personnel.

The parameter trending display shall contain recent and current magnitudes of the parameter as a function of time. The derivation and presentation of parameter trending during upset conditions is a task that may be automated, thus freeing the operator to interpret the trends rather than generate them. Display of time derivatives of the parameters in lieu of trends to both optimize operator-process communication and conserve space may be acceptable.

The SPDS may be a source of information to other systems, and the functional criteria of these systems shall state the required interfaces with the SPDS. Any interface between the SPDS and a safety system shall be isolated in accordance with the safety system criteria to preserve channel independence and ensure the integrity of the safety system in the case of SPDS malfunction. Design provisions shall be included in the interfaces between the SPDS and nonsafety systems to ensure the integrity of the SPDS upon failure of nonsafety equipment.

A qualification program shall be established to demonstrate SPDS conformance to the functional criteria of this [NUREG-0696] document.

18.1.17.2 Union Electric Response

Supplement 1 to NUREG-0737 (Generic Letter 82-33, dated December 17, 1982) provided guidance and requirements for the SPDS which superseded previous NRC guidance.

A detailed description of the SPDS conceptual design, along with some of the details of the Emergency Response Facility Information System, was provided to the NRC in SLNRC 81-38 dated June 1, 1981. Further details were provided to the NRC in SLNRC 83-19, dated April 15, 1983. As stated in that reference, the SPDS was required to be fully operational prior to start-up from the first refueling. SLNRC 84-03, dated January 13, 1984 submitted a safety analysis for the SPDS to NRC per the requirements of GL 82-33. The SPDS verification and validation program completion was documented in SLNRC 84-111; dated September 5, 1984 and SLNRC 86-4, dated February 27, 1986. In addition, ULNRC-1288, dated April 8, 1986 was written to close out GL 82-33 and inform NRC that the emergency response facilities and information system (which includes the SPDS) were complete.

The Callaway SPDS was accepted by the NRC by letter dated August 3, 1987 - T. W. Alexion to D. F. Schnell. (In addition, an NRC Inspection (NRC Inspection Report No. 50-483/88009, dated July 19, 1988) of the Callaway Emergency Response Facility found no discrepancies or open items associated with the SPDS.)

On April 12, 1989, the NRC issued Generic Letter No. 89-06 which required licensees to certify that the SPDS meets the requirements of NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342. This certification was provided to the NRC via ULNRC-2034, dated July 10, 1989.

SLNRC 83-19 and ULNRC-2034 describe the SPDS display console being located next to main control room consoles RL001/RL002. The SPDS display console is no longer located next to RL001/RL002. The SPDS displays are available on several plant computer terminals located in the Control Room. The plant computer terminal located near RL005/RL006 has been designated as the primary plant computer terminal for SPDS displays since the primary SPDS plant computer terminal previously located next to RL001/RL002 has been eliminated.

18.1.17.3 Conclusions

Based on the above summary of SPDS activities, the UE SPDS design agrees adequately with the requirements established in NUREG-0737, Supplement 1.

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

Partial natural circulation testing has been conducted at Callaway (on 10/14/84) to insure and confirm 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 demonstrated the following plant characteristics: Length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation. The simulator has full capability of simulating natural circulation and utilizes actual plant data.

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

The Union Electric natural circulation testing requirements are based on post-Three Mile Island (TMI) regulatory positions and NRC Branch Technical Position RSB 5-1. In response to the post-TMI positions, Westinghouse developed a special low-power test program which was approved by NRC. In accordance with the test program, prototype natural circulation testing for 4-loop plants was performed at the Diablo Canyon Plant. The prototype testing was performed in 1985 and acceptably demonstrated plant characteristics important to plant shutdown under natural circulation conditions, such as, length of the time to achieve natural circulation, the ability to borate the reactor coolant system under natural circulation conditions and the ability to cool the plant down and depressurize under natural circulation conditions. The Diablo Canyon test results have been determined to be applicable to the Callaway design.

18.1.18.3 Conclusion

The Union Electric testing (ETT-ZZ-09240) and training program meets Item I.G.1 of NUREG-0737.

18.2 SITING AND DESIGN

18.2.1 POST-ACCIDENT REACTOR COOLANT SYSTEM VENTING (II.B.1)

18.2.1.1 NRC Guidance Per NUREG-0737

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that ensures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Clarification

A. General

(1) The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

(2) Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the

normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

(3) The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

(4) Where practical, the reactor coolant system vents should be kept smaller than the size corresponding to the definition of LOCA (10 CFR 50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series with a vent valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

(5) A positive indication of valve position should be provided in the control room.

(6) The reactor coolant vent system shall be operable from the control room.

(7) Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for the reactor pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures may be a viable option to meet the single-failure criterion. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

(8) The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system, when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

(9) Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

(10) The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344-1975 as supplemented by Regulatory Guide 1.100, 1.92 and SEP 3.92, 3.43, and 3.10. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

(11) Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

(12) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) Integration into emergency procedures.

(c) Integration into operator training.

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

B. PWR Vent Design Considerations

(1) Each PWR licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting noncondensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

(2) Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that ensures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

(3) Venting of the pressurizer is required to ensure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

18.2.1.2 Union Electric Response

The Callaway design provides the capability of venting the RCS to ensure that, if noncondensible gases become present in the RCS, regardless of the means postulated for generation of such noncondensibles, gases can be vented from the system, thereby ensuring that the flow paths associated with natural circulation core cooling capability are maintained. The venting capability is provided by the existing redundant pressurizer power-operated relief valves (PORVs) and their associated motor-operated isolation valves which can be used for the venting of the pressurizer and by the reactor vessel head vent system which provides redundant venting capability of the reactor vessel, RCS hot leg piping, and RCS cold leg piping via bypass leakage paths to the vessel head. The design features of these systems are discussed below.

The capability for venting of the pressurizer and the reactor vessel head is provided via safety grade, Class 1E, environmentally qualified, seismic Category I, redundant systems, which meet the single failure criteria assuring both vent opening and vent closing capabilities. Block valves are an integral part of both the pressurizer and reactor vessel head vent system and meet the same qualification requirements as the vent valves.

The size of the RCS vents is determined as follows:

1. The pressurizer vent was based on the existing PORV (3-inch valve) capabilities.

2. The reactor vessel head vent system incorporates a 3/8-inch orifice to limit the maximum reactor coolant flow rate to a value less than that which defines a LOCA (see Figure 18.2-1).

The design provides for a motor-operated isolation valve in series with each pressurizer PORV. These PORV isolation valves are remotely actuated from the control room. Control room indication is provided for the pressurizer PORVs and PORV isolation valves and for the reactor vessel head vent valves. Each vent is remotely operable from the control room. An individual handswitch is provided for each valve.

The design of the RCS venting systems minimizes the probability of an inadvertent opening and consequence of such an opening.

1. The pressurizer vent system:

   The pressurizer PORVs are normally closed, Class 1E solenoid valves that energize to open. Thus, loss of power will not actuate these valves. The

PORV isolation valves are normally open, motor-operated valves. Assuming an inadvertent opening of the PORV or its failure to close, operator action is taken to close the associated block valve.

2. The reactor vessel head vent system:

Each of the redundant vent paths off of the reactor vessel head contains two in-series, normally closed, same safety train, Class 1E, environmentally qualified solenoid valves. The two normally closed valves in series limit any postulated events which could result in an inadvertent opening of the vent.

The pressurizer will vent to the pressurizer relief tank. The reactor vessel head vent system valves are located on the Integrated Head Assembly walkway above the reactor vessel. The discharge from these valves will be directed to the open area of the containment above the refueling pool. This area precludes the potential for forming stagnant pockets of vented gases. Mixing and cooling of the vented gases will be accomplished using permanent plant systems.

The Westinghouse Owners Group (WOG) developed a generic reactor vessel head vent guideline. The SNUPPS utilities used the generic guidance developed by the WOG in the development of procedures for use of the original head vent system. This generic guidance is also applied for use of the new head vent system provided by AREVA, which is of a similar design.

Testing of Category B valves is performed in accordance with subsection ISTC of the ASME Code for Operation and Maintenance of Nuclear Power Plants as allowed by 10 CFR 50.55a.

18.2.1.3 Conclusion

The Callaway design for the postaccident reactor coolant system vent system meets the applicable requirements of item II.B.1 of NUREG-0737.

18.2.2 DESIGN REVIEW OF THE PLANT SHIELDING (II.B.2)

18.2.2.1 NRC Guidance Per NUREG-0737

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the

location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Clarification

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. These actions should be taken pending conclusions resulting in the long-term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station, and sample analysis area must be included among those areas where access is considered vital after an accident. (See Item III.A.1.2 for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to ensure that personnel can perform the necessary postaccident operations in the vital areas, the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

(1) Source Term

The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., you may assume that the radioactive decay that occurs before fission products can be transported to various systems).

(a) Liquid-Containing Systems: 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal (RHR), high-pressure coolant injection (HPCI), and low-pressure coolant injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, you may assume that the water contains no noble gases.

(b) Gas-Containing Systems: 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines connected to the primary system (e.g., BWR steam lines), the concentration of radioactivity shall be determined, assuming that the activity is contained in the vapor space in the primary coolant system.

(2) Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system (CVCS), containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from the leakage of systems located outside of the containment need not be considered for this analysis. Leakage measurement and reduction is treated under Item III.D.1.1, "Integrity of Systems Outside Containment Likely To Contain Radioactive Material for PWRs and BWRs." Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Item III.D.1.4, "Radwaste System Design Features To Aid in Accident Recovery and Decontamination."

(3) Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the

accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case bases. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident.

(a) Areas Requiring Continuous Occupancy: <15 mrem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

(b) Areas Requiring Infrequent Access: GDC 19. These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/ chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

(4) Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during postaccident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

(a) LOCA events which completely depressurize the primary system should consider releases of the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to the containment atmosphere.

(b) LOCA events in which the primary system may not depressurize should consider the source term (100 percent noble gases, 50

percent iodines, and 1 percent particulate) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of the containment. Non-LOCA events both inside and outside of the containment should use 10 percent noble gases, 10 percent iodines, and 0 percent particulate as a source term.

The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/ Particulate)
	%	%
Outside	(100/50/1) in RCS	(10/10/0) in RCS
Inside	Larger of (100/50/1) in containment or (100/50/1) in RCS	(10/10/0) (in RCS

18.2.2.2 Union Electric Response

The shielding design criteria used for the SNUPPS plants is in accordance with NRC Standard Review Plan 12.2 and is described in Section 12.3.2 of the FSAR. Two basic plant conditions are the bases of the shielding design, normal full power operation, and plant shutdown. The shielding design objectives for these conditions and anticipated operational occurrences, as stated in Section 12.3.2.1, are:

a. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and proximate site boundary occupants are ALARA and within the limits of 10 CFR 20.

b. To ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

c. To reduce potential equipment neutron activation and mitigate the possibility of radiation damage to materials.

d. The control room will be sufficiently shielded, so that the direct dose plus the inhalation dose (calculated in Chapter 15.0) will not exceed the limits of GDC-19.

Radiation zones have been established, based on required personnel access during these plant conditions.

18.2.2.2.1 Design Review of Plant Shielding

18.2.2.2.1.1 General

The following discussion provides a description of the design review of plant shielding of spaces around systems that may contain highly radioactive materials as a result of an accident. Systems required to process reactor coolant outside the containment during post-accident conditions were selected for evaluation.

The radiation and shielding design review was performed to identify the location of vital areas and equipment such as the control room, sample station, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, the review results ensure that adequate access to vital areas and protection of safety-related equipment are provided.

As shown in Figures 18.2-2 through 18.2-11, a number of radiation zone maps and associated dose rate decay curves have been produced as a result of the design review. Radiation levels for various areas around contaminated systems for various times can be found on these maps and curves. Operators may refer to the maps and curves to apprise themselves of the locations of potentially high radiation areas for any time following the postulated accident. These maps and curves will be available in the critical post-accident control and support areas, e.g., control room, TSC, etc., for use following postulated DBA's.

18.2.2.2.1.2 Scope of Design Review

18.2.2.2.1.2.1 Systems Engineering Methodology

A. Selection of Systems for Shielding Review

Plant systems considered in the shielding review are classified into the following categories:

Category A (Recirculation Systems)

The first category of systems are those systems designed to mitigate a design basis loss of coolant accident and which might contain highly

radioactive sources. Such systems include the emergency core cooling systems.

For the shielding review, the ECCS systems were postulated to contain significant additional sources of radioactivity in excess of the original plant design basis.

The following systems have been selected to ensure the radiation safety concern is adequately addressed by the existing plant shielding design:

(1) Those portions of the containment spray systems used to recirculate water from the containment sump back into the containment.

(2) Those portions of the residual heat removal systems used to recirculate water from the containment sump back into the containment.

(3) Those portions of the safety injection system used to recirculate water from the containment sump back into the containment.

(4) Those portions of the chemical and volume control system (CVCS) used to recirculate water from the containment sump back into the containment.

Category B (Extensions of Containment Atmosphere)

The second category of systems are the systems or portions of systems which would contain radioactivity by virtue of their connection to the containment atmosphere following an accident. These systems would not be expected to contain a significant level of radioactive sources that are considered in this shielding review, since proper operation of the emergency core cooling systems is expected to prevent extensive core damage. Nevertheless, such sources have been postulated in the following system:

(1) Those portions of the post-accident containment hydrogen analyzer system external to the containment which would contain the atmosphere from the containment.

(2) Those portions of the post-accident sampling system used to obtain a containment atmosphere sample.

Category C (Liquid Samples)

The third category of systems is sampling systems. As discussed in Section 18.2.3, NUREG-0737, Task II.B.3 requires that certain

post-accident liquid samples be obtained from the reactor coolant system or containment systems. Those portions of the sampling system which must be used to meet the intent of Task II.B.3 were included in this shielding review.

B. Radioactive Source Release Fractions

Per NUREG-0737, the following release fractions were used as a basis for determining the concentrations for the shielding review:

(1) Source A: Containment atmosphere - 100 percent noble gases, 25 percent halogens

(2) Source B: Reactor coolant - 100 percent noble gases, 50 percent halogens, 1 percent solids

(3) Source C: Containment sump liquid - 50 percent halogens, 1 percent solids

These release fractions were applied to the total curies available for the particular chemical species (i.e., noble gas, halogens, or solid) for an equilibrium fission product inventory for the SNUPPS core.

The release fraction for Cs was assumed to be 1 percent for the purposes of this shielding review. However, a relationship was developed which related the dose rates calculated, assuming 1 percent Cs, to the dose rate that would be expected if 50 percent of the Cs was released to the liquid source (as recommended by Revision 1 to Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants"). This relationship is provided in Figure 18.2-11. No noble gases were included in the containment sump liquid (Source C) because Regulatory Guide 1.7 has set this precedent in modeling liquids in the containment sump.

C. Source Term Models

The preceding section (B) outlines the assumptions used for release fractions for the shielding design review. However, these release fractions are only the first step in modeling the source terms for the activity concentrations in the systems under review. The important modeling parameters, decay time, and dilution volume obviously also affect any shielding analysis. The following sections outline the rationale for the selection of values for these key parameters.

(1) Decay Time

For the first stage of the shielding design review process, no decay time credit was used with the above releases. The primary reason for this was to develop a set of normalized accident radiation zone maps (i.e., no decay) that could be used as a tool by the plant staff along with a set of decay curves to quantitatively assess the plant status quickly following any abnormal occurrence. Decay curves are provided for the containment atmosphere and containment sump liquid only. Except for areas adjacent to the containment, the sump liquid source will be the dominating contributor.

(2) Dilution Volume

The volume used for dilution is important, since it affects the calculations of dose rate in a linear fashion. The following dilution volumes were used with the release fractions and decay times listed above to arrive at the actual source terms used in the shielding reviews:

(a) Source A: Containment free volume.

(b) Source B: Reactor coolant system volume.

(c) Source C: The volume of water present at the time of recirculation (reactor coolant system + refueling water storage tank + accumulator tanks).

(3) Associated Sources and Systems

For the following systems, the source considered is listed. Note that normally shut valves were assumed to remain shut.

(a) Containment spray system - At the initiation of recirculation, Source C was used.

(b) Safety injection system - At the initiation of recirculation, Source C was used.

(c) Residual heat removal system - Source C was used for sump recirculation mode.

(d) Sampling system - The sources used in the shielding design review for sampling systems were as follows:

Containment air sample - Source A
Reactor coolant sample - Source B
Containment sump sample - Source C

(e) CVCS system - The liquid source was Source C.

18.2.2.2.1.2.2 Shielding Design Review Methodology

A. Analytical Shielding Techniques

The previous sections outlined the rationale and assumptions used for the selection of the systems in the shielding design review, as well as the formulation of the sources for those systems. The next step in the review process was to use those sources to estimate dose rates from those selected systems. The dose rates were determined using a point-kernel computer code developed by Bechtel. This code utilizes the semi-emperical methods developed by Rockwell (Reference 8) for calculating the direct gamma dose rates. To determine the dose rate contribution from the containment, QAD-CG (Reference 9) was used. For corridors outside compartments, reviews were done to check the dose rate transmitted into the corridor through the walls of adjacent compartments. Checks were also made for any piping or equipment that could directly contribute to corridor dose rates, i.e., piping that may be running directly in the corridor or equipment/piping in a compartment that could shine directly into corridors with no attenuation through compartment walls.

B. Accident Radiation Zone Maps

Radiation levels are evaluated using the radiation zone maps, Figures 18.2-2 through 18.2-9, and associated decay curves, Figures 18.2-10 and 18.2-11, and are considered in parallel with required operator actions.

The zone boundaries were formulated based on the following rationale:

Zone Designation	Rationale	D, Zone Dose Rate Limits (Rem/hr)
A-I	The first zone is consistent with the personnel radiation exposure guidelines of Task II.B.2 of NUREG-0737 for vital areas.	$0 \leq \dot{D} \leq 0.015$

Zone Designation	Rationale	D, Zone Dose Rate Limits (Rem/hr)
A-II	The second zone is consistent with the personnel radiation exposure guidelines of Task II.B.2 of NUREG-0737 for vital areas requiring infrequent access or corridors to these areas. Such zones involve no time and motion evaluations.	$0.015 \leq \dot{D} \leq 0.100$
A-III	The third zone is consistent with the personnel radiation exposure guidelines of Task II.B.2 of NUREG-0737. Zones in this range required that a time and motion study be done to ensure that integrated exposure was not greater than 5 Rem as given in General Design Criteria 19.	$0.100 < \dot{D} \leq 5$
A-IV		$5 < \dot{D} \leq 50$
A-V		$50 < \dot{D} \leq 500$
A-VI		$500 < \dot{D} \leq 5000$
A-VII		$5000 < \dot{D} \leq 50{,}000$
A-VIII		$50{,}000 < \dot{D} \leq 500{,}000$

18.2.2.2.1.2.3 Personnel Exposure Limits and Methodology

A. Access

Operator actions that are required post-LOCA were reviewed to ensure that first priority safety actions can be achieved in the postulated radiation fields. This review ensures that access is available and required operator actions can be achieved as discussed in Section 18.2.2.2.1.3.

B. Personnel Radiation Exposure Guidelines

The general basis for personnel radiation exposure guidelines was 10 CFR 50, Appendix A, GDC 19. The following additional radiation limit guidelines were used to evaluate occupancy and accessibility of plant vital areas. General area dose rates were used rather than maximum surface dose rates. Contributions from all sources were considered.

(1) Vital areas requiring continuous occupancy

Vital areas such as control room, counting room, laboratory, and the onsite technical support center were verified to ensure the direct dose rate was less than 15 mr/hr. The 30 day average direct radiation dose rate is less than 15 mr/hr for the SAS room and the control room toilet.

(2) Vital areas requiring infrequent access or corridors to these vital areas

For these areas, the dose rate was verified to be less than 5 R/hr except as noted in Section 18.2.2.2.1.3.

18.2.2.2.1.3 Results of Review

The shielding design criteria and objectives have been met in the design of the Callaway Plant. These criteria and objectives have been extended to the areas designated to be the onsite Technical Support Center and the Operations Support Center, as required by the expected occupancy of these areas. The following is a discussion of the impact of a postulated LOCA or TMI-2 type event on the shielding design and is based on the Callaway specific system design capabilities:

A. LOCA

Assuming a DBA LOCA with radiation source terms consistent with Regulatory Guides 1.4 and 1.7, plus the Cs fraction discussed in Section 18.2.2.2.1.2.1, all safety-related equipment and instrumentation will be qualified for the maximum equipment doses associated with the time that the equipment must function. All safety-related systems operations are performed either automatically or remote manually from the control room. Operations within the auxiliary building are not expected following a LOCA. During the long-term recovery phase, access to sample stations in the auxiliary building may be limited. As discussed above, the dose limitations of GDC-19 for control room operators are met.

B. TMI-2

Callaway is designed to preclude events similar to the TMI-2 event. For example, the Callaway design includes reactor coolant system high point

vents (as discussed in Section 18.2.1) and the associated Class 1E instrumentation required to detect inadequate core cooling and thus precludes the degradation of the fuel cladding and any massive release of activity to the coolant. However, assuming that a TMI-2 event does occur, contamination of the auxiliary building is precluded by design:
1) Compliance with containment isolation criteria is described in Section 6.2.4 and Section 18.2.11 and precludes contamination of the auxiliary building by auxiliary systems, and 2) the Callaway design includes a dedicated, safety-related letdown system located totally within the containment which provides controlled letdown capability to the pressurizer relief tank, eliminating any operational need to contaminate the chemical and volume control system in the auxiliary building.

Based on the above, dose rates were not evaluated in the auxiliary building for an undiluted reactor coolant system source term being present in the residual heat removal system. Dose rates inside containment due to the TMI-2 type event have been considered for equipment qualification. Habitability of the TSC is addressed in Section 18.3.2.2.

18.2.2.3 Conclusion

The shielding design criteria and objectives for the Callaway Plant meet the applicable recommendations of item II.B.2 of NUREG-0737. Radiation qualification of safety-related equipment is addressed in Section 3.11(B).

18.2.3 POSTACCIDENT SAMPLING SYSTEM (II.B.3)

18.2.3.1 NRC Guidance Per NUREG-0737

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 Rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release.

The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses, assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Clarification

The following items are clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13 and October 30, 1979 clarification letters.

(1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

(2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

(a) Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes).

(b) Hydrogen levels in the containment atmosphere.

(c) Dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

(3) Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

(4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant

samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

(5) The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions, the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

(6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). [Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).]

(7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants.)

(8) If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

(9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 mCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources, such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

(10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.

(11) In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

(c) Guidelines for analytical or instrumentation range are given below in Table II.B.3-1.

18.2.3.2 Union Electric Response

WCAP-14986-A, Revision 2, ‘Post Accident Sampling System Requirements: A Technical Basis,’ evaluated the post accident sampling system (PASS) requirements to determine their contribution to plant safety and accident recovery. The topical report considered the progression and consequences of core damage accidents and assessed the accident progression with respect to plant abnormal and emergency operating procedures, severe accident management guidance, and emergency plans. WCAP-14986 concluded that the current PASS samples specified in NUREG-0737, ‘Clarification of TMI Action Plan Requirements,’ may be eliminated (i.e., remove the requirements to perform the sampling from the licensing basis).

The NRC issued a safety evaluation dated June 14, 2000 approving WCAP-14986 with additional licensee required actions. Callaway Plant License amendment [144] dated [April 6, 2001] implemented WCAP-14986 and the associated NRC safety evaluation dated June 14, 2000, which removed the program requirements of Technical Specification Section 5.5.3, "Post Accident Sampling."

18.2.3.3 Conclusion

Based on the above discussion, item II.B.3 of NUREG-0737 is no longer applicable to the Callaway Plant.

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

18.2.4.1 NRC Guidance Per NUREG-0737

Position

The staff requires 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 primary 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_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

(2) H_2 flammability and explosive limit, sources of O_2 in containment or reactor coolant system.

18.2.4.2 Union Electric Response

Union Electric presents a course on mitigating core damage to licensed operators which covers the topics outlined in section 18.2.4.1. Training of Callaway Plant personnel to recognize and mitigate the consequences of core damage meets the intent of the Institute of Nuclear Power Operations Guidelines, Rev. 1, 1/15/81.

18.2.4.3 Conclusion

Union Electric's training program for mitigating core damage satisfies NUREG-0737.

18.2.5 PERFORMANCE TESTING OF THE PRESSURIZER POWER-OPERATED RELIEF VALVE (II.D.1)

18.2.5.1 Background

The Report of the President's Commission on the Accident at Three Mile Island, Findings A.3 and A.4, describes the role the failure to close the power-operated relief valve (PORV) had in the resulting accident. The Report found that failure of the valve initiated the accident, but also found that the operating crew and utility management failed to diagnose the occurrence and consequences of the PORV failure. This latter item contributed more to the consequences of the accident than the PORV failure to close.

The NRC, in its review of the accident at TMI-2, concluded that additional assurance should be provided that PORVs and safety valves will perform as designed and that indication of the status of these valves must also be provided in the control room. The first item is discussed below. The second item is discussed in Section 18.2.6.

18.2.5.2 NRC Guidance Per NUREG-0737

Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981 for BWRs and July 1, 1982 for PWR's:

    (1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation of other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions, as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data, including criteria for success and failure of valves tested, must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

B. Qualification of PWR Block Valves--Although not specifically listed as a short-term lessons learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under task item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the generic EPRI PWR safety and relief valve testing program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions.

C. ATWS Testing--Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3,200 to 3,500 (Service Level C pressure limit) psi and 700°F with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized-water reactors.

18.2.5.3 Discussion

The PORVs in the Callaway design are relied on to function to alleviate overpressurization that possibly could occur during startup of the reactor, during cold shutdown conditions, and they may be relied on to function during shut down of the reactor, assuming only safety-grade equipment is functioning. (These functions are described in Sections 5.2 and 5.4(A).) The PORVs are required to function to mitigate the consequences of the inadvertent ECCS actuation at power event.

The PORVs are also designed to limit high pressure during normal operation. The description of this control function is presented in Sections 5.2 and 7.6. As discussed

below, operability of the PORVs will be demonstrated by prototypical testing and appropriate analyses.

The safety valves for the Callaway design are relied on to limit primary system pressure following anticipated operational transients. The design basis for the safety valves is presented in Section 5.2. The valves are required by ASME Boiler and Pressure Vessel Code to mitigate excessive pressure increases, regardless of their source. As discussed below, operability of the safety valves was demonstrated by prototypical testing and appropriate analyses.

18.2.5.4 Union Electric Response

The reactor coolant system is provided with two PORVs and three code safety valves. Each PORV also has an associated motor-operated block valve.

The PORVs for Callaway were manufactured by Crosby; the safety valves were manufactured by Crosby. These valves are included in the safety and relief valve testing program that has been developed by EPRI. A description of this program entitled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," dated December 13, 1979, was submitted to the NRC on December 17, 1979 (letters from W. J. Cahill, Jr., Chairman of EPRI Safety and Analysis Task Force, to H. Denton and D. Eisenhut, NRC). A revision to this program was submitted to the NRC in July 1980. The NRC staff completed its review of this program and found it acceptable.

An interim report on these valve tests was submitted by the PWR utilities to the NRC in July 1981. A final report on these tests was provided in Reference 12.

Preoperational testing of the PORVs includes monitoring the dynamic response of the relief valve discharge piping during actuation of the PORVs. These in-plant dynamic tests will be initiated with a water-solid inlet (loop seal) at the PORVs and a steam bubble maintained in the pressurizer.

Regarding verification of the block valve functionability, Callaway qualification of the block valves was provided in the letter SLNRC 82-030 on July 1, 1982.

Based on the NRC review of the submittals addressing safety valves, PORVs, PORV block valves and associated piping, Union Electric was requested to provide additional information. This information was provided by SNUPPS letters dated June 30, 1986 (SLNRC 86-07) and September 26, 1986 (SLNRC 86-09). Additional supporting information was transmitted to NRC by ULNRC-1500 (dated April 24, 1987) and ULNRC-1517 (dated May 29, 1987). In a September 10, 1987 transmittal, NRC formally accepted Union Electric's response to NUREG-0737, II.D.1 contingent upon UE committing to a valve inspection after each lift involving seal water discharge. ULNRC-1681 (dated November 17, 1987) committed to either replace or inspect a valve, after lift, prior to placing the plant back in service.

18.2.5.5 Conclusion

The Callaway plan to demonstrate the operability of the PORVs and safety valves satisfies the guidance of item II.D.1 in NUREG-0737, as discussed in the NRC Safety Evaluation Report dated September 10, 1987.

18.2.6 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (II.D.3)

18.2.6.1 NRC Requirement Per NUREG-0737

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Clarification

(1) The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

(2) The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

(3) The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single-channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis of an action.

(4) The valve position indication should be seismically qualified, consistent with the component or system to which it is attached.

(5) The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift) and in accordance with Commission Order, May 23, 1980 (CLI-20-81).

(6) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

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

(b) Integration into emergency procedures.

(c) Integration into operator training.

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

18.2.6.2 Union Electric Response

Safety-grade position indication is provided for each safety valve and power-operated relief valve (PORV) that indicates when the valve is not in its fully closed position. The position indication is seismically and environmentally qualified. The position indication for each valve is displayed in the control room, and an alarm is provided if any of the PORVs or safety valves is not fully closed.

Other, nonsafety-related instrumentation is provided on the valve discharge piping and the pressurizer relief tank to provide an alternate means of assessing the status of the safety valves and PORVs (see Figure 5.1-1, Sheet 2).

18.2.6.3 Conclusion

The Callaway design satisfies the guidance of Item II.D.3 of NUREG-0737.

18.2.7 AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION (II.E.1.1)

18.2.7.1 Background

The initial response of the auxiliary feedwater system (AFS) at TMI-2 was interrupted by two closed block valves; one valve in each auxiliary feedwater train. The closed valves prevented feedwater from reaching the steam generators when the main feedwater system pumps tripped off. The Report of the President's Commission on the Accident at Three Mile Island, Finding E.5.b states:

> "There were deficiencies in the review, approval, and implementation of TMI-2 plant procedures."

More specifically:

"(vi) Performance of surveillance tests was not adequately verified to be sure that the procedures were followed correctly. On the day of the accident, emergency feedwater block valves which should have been open were closed. They may have been left closed during a surveillance test 2 days earlier."

However, the Report did not find that the isolation of the auxiliary feedwater system was a pivotal event in the accident sequence. Since the B&W design included provisions to remove decay heat and ensure core cooling without auxiliary feedwater, the total failure of the nonsafety grade auxiliary feedwater system at TMI-2 was in fact a design basis for the design of emergency safety systems.

The NRC in its review of the accident assigned more significance to the failure of the auxiliary feedwater system. The NRC concluded that additional evaluation and requirements should be placed on the auxiliary feedwater system. These items are discussed in NUREG-0737 and are presented below.

18.2.7.2 NRC Guidance per NUREG-0737

Position - AFS Evaluation

The office of Nuclear Reactor Regulation is requiring re-evaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages.

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance.

(3) Reevaluate the AFW system flowrate design bases and criteria.

Clarification - AFS Evaluation

Operating License Applicants - Operating license applicants have been requested to respond to staff letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W). These responses will be reviewed during the normal review process for these applications.

18.2.7.3 Union Electric Response

A reliability analysis of the auxiliary feedwater system (AFS) was submitted to the NRC by letter SLNRC 81-44, dated June 8, 1981. A comparison of the design with Standard Review Plan 10.4.9 and Branch Technical Position ASB 10-1 is provided in Section 10.4.9. An evaluation of the auxiliary feedwater system flowrate design bases and criteria was submitted by letter SLNRC 81-39, dated June 3, 1981. This reliability

analysis and its assumptions on AFS pump availability is separate from the deterministic analyses of Section 15.2 (See Callaway Amendment No. 168).

The NRC staff reviewed the SNUPPS AFS design capabilities against the recommendations of the March 10, 1980 NRC letter (D. Ross, NRC to all pending W and C-E License Applicants) which corresponds to NUREG-0737, Item II.E.1.1. Based on this review, a confirmatory licensing issue was identified, regarding physically securing the condensate storage tank manual isolation valve. This issue was resolved prior to initial fuel load in Callaway SSER No. 3 (NUREG-0830, dated May, 1984).

18.2.7.4 Conclusion

The Callaway design and analyses for the AFS meets the recommendations of Item II.E.1.1 of NUREG-0737.

18.2.8 AUXILIARY FEEDWATER INITIATION AND INDICATION (II.E.1.2)

18.2.8.1 NRC Guidance Per NUREG-0737

Position - AFS Automatic Initiation

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFS.

(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFS function.

(3) Testability of the initiation signals and circuits shall be a feature of the design.

(4) The initiating signals and circuits shall be powered from the emergency buses.

(5) Manual capability to initiate the AFS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Clarification - AFS Automatic Initiation

The intent of this recommendation is to ensure a reliable automatic initiation system. This objective can be met by providing a system which meets all the requirements of IEEE Standard 279-1971.

Position - AFS Flowrate Indication

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain the actual performance of the AFS when it is called to perform its intended function, the following requirements shall be implemented:

(1) Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

(2) The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Clarification - AFS Flowrate Indication

The intent of this recommendation is to ensure a reliable indication of AFS performance. This objective can be met by providing an overall indication system that meets the following appropriate design principles:

For Westinghouse and Combustion Engineering Plants

(1) To satisfy these requirements, W and C-E plants must provide as a minimum one auxiliary feedwater flowrate indicator and one wide-range steam-generator level indicator for each steam generator or two flowrate indicators.

(2) The flow indication system should be:

(a) Environmentally qualified

(b) Powered from highly reliable, battery-backed non-Class 1E power source

(c) Periodically testable

(d) Part of plant quality assurance program

(e) Capable of display on command

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed, taking into consideration:

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

b. Integration into emergency procedures.

c. Integration into operator training.

d. Other alarms during emergency and need for prioritization of alarms.

18.2.8.2 Union Electric Response

Automatic initiation of the AFS meets the NRC recommendations as described in Sections 10.4.9 and 7.3.6. The AFS flowrate indication meets the NRC recommendations, as described in Section 10.4.9 and 7.5.

The NRC staff reviewed the SNUPPS AFS design capabilities against the recommendations of the March 10, 1980 NRC letter (D.Ross, NRC to all pending W and C-E License Applicants) which corresponds to NUREG-0737, Item II.E.1.2. Based on this review, a confirmatory licensing issue was identified, regarding physically securing the condensate storage tank manual isolation valve. This issue was resolved prior to initial fuel load in Callaway SSER No. 3 (NUREG-0830, dated May, 1984).

18.2.8.3 Conclusion

The Callaway design and analyses for the AFS meet the recommendations of Item II.E.1.2 of NUREG-0737.

18.2.9 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS (II.E.3.1)

18.2.9.1 Background

The Report to the President on the Accident at Three Mile Island, in the Account of the Accident, speaks of the inability of the operators at TMI-2 to establish core cooling prior to gross fuel damage. The Report does not conclude that the pressurizer heaters were required to establish core cooling or that they are required for natural circulation. The Technical Staff Analysis Report, "Summary of Sequence of Events," Appendix B, "Significant Equipment Problems," states that the operators experienced "equipment problems that may have drawn the operators' attention away from those principal actions necessary to protect the reactor core." In particular, "throughout the sequence, the operators experienced trouble with the pressurizer heaters tripping. This tripping could be attributed to grounding due to the moisture being injected into the reactor building during the course of the accident."

The NRC included an item in NUREG-0578 and in subsequent TMI-related documents recommending that one of the possible pressurizer heater power supplies include an emergency power source. The NRC's recommendation is presented below.

18.2.9.2 NRC Guidance Per NUREG-0737

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Clarification

(1) Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only Class 1E division power supply.

(2) The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

(3) The power sources need not necessarily have the capacity to provide power to the heaters concurrently with the loads required for loss-of-coolant accident.

(4) Any changeover of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

(5) In establishing procedure to manually load the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

(a) Which ESF loads may be appropriately shed for a given situation.

(b) Reset of the safety injection actuation signal to permit the operation of the heaters.

(c) Instrumentation and criteria for operator use to prevent overloading a diesel generator.

(6) The Class 1E interfaces for main power and control power are to be protected by safety-grade circuit breakers (see also Regulatory Guide 1.75).

(7) Being non-Class 1E loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal (see item 5.b. above)."

18.2.9.3 Union Electric Response

The total capacity of the pressurizer heaters at 480 V ac is 1,799 Kw (Table 5.1-1). The pressurizer heaters are divided into three groups (see Figure 8.3-2). The rated capacity of each group at 480 V ac is as follows:

Group A - 692 Kw
Group B - 692 Kw
Group C - 415 Kw

The group C heaters are used for proportional control during power operation.

Groups A and B are the backup heater groups; each of these two groups is powered from a Class 1E power source. This power is interrupted by the load shedder/sequencer following a safety injection or emergency bus undervoltage signal.

The controls for each backup pressurizer heater group are provided with redundant non-Class 1E ac power sources--one from the 480-Vac system and one from the 125-Vdc system via a 125-Vdc/120-Vac inverter (see Figure 8.3-6). Each battery charger of the 125-Vdc system is supplied from a single separation group of the 4.16-kV onsite emergency distribution system. When the 480-Vac system is unavailable following a loss-of-offsite power, the dc-backed power supplies will supply the backup pressurizer heater controls. Similar to the breakers feeding the heater load centers, the circuit breakers supplying the 125-Vdc battery chargers are automatically tripped upon an SIS or emergency bus undervoltage signal. They may be reclosed from the control room when desired after reset of the breaker tripping signals.

For additional reliability, a cross-tie is provided between Separation Groups 5 and 6 of the non-Class 1E 125-Vdc system. This will permit operation of selected loads of both separation groups in the event of a failure of either battery charger.

All the breakers which function upon SIS and bus undervoltage are seismically qualified isolation devices.

Analysis shows that subcooling would be maintained in the reactor coolant system for up to 4 hours without heat input from the pressurizer heaters. Pressure control for the reactor coolant system, as discussed in Section 5.4(A), can be accomplished without pressurizer heaters. If pressurizer heaters were used for pressure control, analysis indicates that 150 kW is sufficient to maintain subcooling. Plant procedures will be provided for manually connecting (from the control room) pressurizer heaters to emergency power sources following a loss of offsite power.

18.2.9.4 Conclusion

The Callaway design satisfies the guidance of item I.E.3.1 of NUREG-0737.

18.2.10 DEDICATED HYDROGEN PENETRATIONS (II.E.4.1)

18.2.10.1 NRC Guidance Per NUREG-0737

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

Clarification

(1) An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

(2) The dedicated penetration or the combined single-failure proof alternative shall be sized such that the flow requirements for the use of the recombiner or purge system are satisfied. The design shall be based on 10 CFR 50.44 requirements.

(3) Components furnished to satisfy this requirement shall be safety grade.

(4) Licensees that rely on purge systems as the primary means of controlling combustible gases following a loss-of-coolant accident should be aware of the positions taken in SECY-80-399, "Proposed Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations." This proposed rule, published in the Federal Register on October 2, 1980, would require plants that do not have recombiners to have the capacity to install external recombiners by January 1, 1982. (Installed internal recombiners are an acceptable alternative to the above.)

(5) Containment atmosphere dilution (CAD) systems are considered to be purge systems for the purpose of implementing the requirements of this TMI Task Action item.

18.2.10.2 Union Electric Response

The NRC has eliminated the requirement for a postulated hydrogen release associated with a design-basis LOCA from 10 CFR 50.44, and the hydrogen recombiners and purge system discussed below are no longer required.

The postaccident H_2 control is accomplished by redundant hydrogen recombiners which are permanently installed inside the containment. Therefore, dedicated hydrogen control penetrations are not required, and this item is not applicable to the Callaway Plant.

As a backup to the safety-related hydrogen control system, a means of purging hydrogen from the containment is provided. Only the containment penetrations and the associated isolation valves are safety-related in the hydrogen purge system. These penetrations are not the subject of this item, since they do not serve external hydrogen recombiners. Since the hydrogen recombiners are actuated from the control room, the shielding and personnel exposure limitations associated with recombiner use and development of procedures for reduction of doses are not applicable to SNUPPS.

18.2.10.3 Conclusion

Item II.E.4.1 is not applicable to the Callaway Plant.

18.2.11 CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

18.2.11.1 NRC Guidance Per NUREG-0737

Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the re-evaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the

automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position [was to be] enclosed as Attachment 1 [to NUREG-0737].)

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Clarification

(1) The reference to SRP 6.2.4 in position 1 is only to the diversity requirements set forth in that document.

(2) For postaccident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals.

(3) Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981. Requirements for operating plants to review their list of essential and nonessential systems will be issued in conjunction with this guide, including an appropriate time schedule for completion.

(4) Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4.

(5) Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve

basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

(6) The containment pressure history during normal operation should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification. Applicants for an operating license and operating plant licensees that have operated less than one year should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint pressure.

(7) Sealed-closed purge isolation valves shall be under administrative control to ensure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

18.2.11.2 Discussion

The containment isolation system and the containment isolation actuation are described in Sections 6.2.4, 7.3.2, and 7.3.8.

18.2.11.3 Union Electric Response

All lines penetrating the containment are identified in Figure 6.2.4-1. These figures also identify the actuation signal(s) for isolation of those lines requiring isolation. The logic design for containment isolation is such that resetting of the containment isolation signal will not result in the loss of containment isolation. Once the initiating signal is reset, individual valves can be opened from the control room, if required. Reopening of isolation valves is performed on a valve-by-valve or line-by-line basis.

The containment isolation setpoint pressure (Hi-1) that initiates containment isolation (CIS-A) for non-essential penetrations has been reduced to the minimum compatible with normal operating conditions. Refer to SNUPPS letter SLNRC 84-43, dated March 15, 1984. The Technical Specifications establish a limit for containment pressure during normal operations. The Technical Specifications also contain the setpoint for Hi-1 which is based on the normal operation limit and instrument drift and accuracy.

Table 18.2-2 identifies systems as either essential or nonessential. Essential systems are those systems required to have isolation valves open for either safe shutdown or mitigation of the consequences of an accident.

The greatest number of lines are automatically isolated upon initiation of a containment isolation signal, Phase A (CIS-A). A CIS-A is initiated when a safety injection signal (SIS) is initiated. An SIS also initiates a feedwater isolation signal (FWIS) and a steam generator blowdown isolation signal (SGBSIS). The diverse parameters sensed to initiate an SIS are low steam line pressure, low pressurizer pressure, and high containment pressure (Hi-1). The CIS-A logic is shown on Figure 7.2-1, Sheet 8.

The main steam and related lines are automatically isolated upon initiation of a steam line isolation signal (SLIS). The diverse parameters sensed to initiate an SLIS are either low steam line pressure or high negative steam pressure rate and high containment pressure (Hi-2). The SLIS logic is shown on Figure 7.2-1, Sheet 8.

The lines supplying component cooling water to equipment inside the containment is isolated by CIS-B. A CIS-B is initiated by high containment pressure (Hi-3). Diversity for CIS-B is provided in the logic for manual actuation of containment spray, which, when manually actuated, also automatically actuates CIS-B. The CIS-B is shown on Figure 7.2-1, Sheet 8.

The containment purge system is isolated upon initiation of a containment purge isolation signal (CPIS). The diverse parameters sensed to initiate a CPIS are high containment purge exhaust radiation level or a CIS-A signal. The CPIS logic is shown in Figure 7.3-1, Sheet 2.

The guidelines used for post-DBA operability against required pressure differentials of containment mini-purge isolation valves intended for use during plant operation comply with NRC criteria. Documentation of operability was provided by Reference 11. The shutdown purge system isolation valves meet SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves are verified to be closed in accordance with NUREG-0737, Item II.E.4.2.

All containment isolation valves are provided with control switches on the main control board. Manual actuation switches are provided for initiation of CIS-A, SLIS, and CPIS. In addition to diversity, these systems are redundant and meet safety-grade (Class 1E) criteria.

18.2.11.4 Conclusion

The design for the containment isolation system satisfies the requirements of Item II.E.4.2 of NUREG-0737.

18.2.12 ACCIDENT MONITORING INSTRUMENTATION (II.F.1)

18.2.12.1 NRC Guidance Per NUREG-0737

Introduction

Item II.F.1 of NUREG-0660 contains the following subparts:

(1) Noble gas effluent radiological monitor.

(2) Provisions for continuous sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position).

(3) Containment high-range radiation monitor.

(4) Containment pressure monitor.

(5) Containment water level monitor.

(6) Containment hydrogen concentration monitor.

NUREG-0578 provided the basic requirements associated with items 1 through 3 above. NRC staff letters issued to All Operating Nuclear Power Plants dated September 13, 1979 and October 30, 1979 provided clarification of staff requirements associated with items 1 through 6 above. Attachments 1 through 6 present the staff position on these matters.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human factors analysis should be performed (see NUREG-0737, Section II.D.2), taking into consideration:

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

b. integration into emergency procedures,

c. integration into operator training,

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

Attachment 1 Noble Gas Effluent Monitor

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal conditions (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of 10.

Clarification

(1) Licensees shall provide continuous monitoring of high-level, post-accident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in Table II.F.1-1 [of NUREG-0737, presented below]. Typical plant effluent pathways to be monitored are also given in the table.

(2) The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

(3) Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

(4) Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

The design description shall include the following information.

(a) System description, including:

(i) Instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable.

(ii) Monitoring locations (or points of sampling), including description of methods used to ensure representative measurements and background correction.

(iii) Location of instrument readout(s) and method of recording including description of the method or procedure for transmitting or disseminating the information or data.

(iv) Assurance of the capability to obtain readings at least every 15 minutes during and following an accident.

(v) The source of power to be used.

(b) Description of procedures or calculational methods to be used for converting instrument readings to release rate per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

TABLE II.F.1-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENT	Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.
PURPOSE	To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

Design Basis Maximum Range

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detectors (Note: 1 R/hr at 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

10^5 μCi/cc	Undiluted containment exhaust gases (e.g., PWR reactor building purge, BWR drywell purge through the standby gas treatment system). Undiluted PWR condenser air removal system exhaust.

10^4 μCi/cc	Diluted containment exhaust gases (e.g., >10:1 dilution, as with auxiliary building exhaust air).
	BWR reactor building (secondary containment) exhaust air.
	PWR secondary containment exhaust air.
10^3 μCi/cc	Buildings with systems containing primary coolant or primary coolant offgases (e.g. PWR auxiliary building, BWR turbine buildings).
	PWR steam safety valve discharge, atmospheric steam dump valve discharge.
10^2 μCi/cc	Other release points (e.g., radwaste building, fuel handling/ storage buildings).
REDUNDANCY	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	(None) Sampling design criteria per ANSI N13.1.
POWER SUPPLY	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	Continuous and recording as equivalent Xe-133 concentrations or μCi/cc or actual noble gases.
QUALIFICATION	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	Offline monitoring is acceptable for all ranges of noble gas concentrations.
	Inline (induct) sensors are acceptable for 10^2 μCi/cc to 10^5 μCi/cc noble gases. For less than 10^2 μCi/cc, offline monitoring is recommended.
	Upstream filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident.

For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.

Attachment 2 Sampling of Plant Effluents

Sampling of Plant Effluents

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

Clarification

(1) Licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of the enclosed Table II.F.1-2 [from NUREG-0737, presented below]. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize noble gas interference.

(2) The shielding design basis is given in Table II.F.1-2 [of NUREG-0737]. The sampling system design shall be such that plant personnel could remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC-19 of 5-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

(3) The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ±20 percent.

Further departure from the isokinetic condition need not be considered in design. Corrections for nonisokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969, may be considered on an ad hoc basis.

(4) Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

TABLE II.F.1-2

SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH-RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT	-	Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.
PURPOSE	-	To determine quantitative release of radioiodines and particulates for dose calculation and assessment.
DESIGN BASIS SHIELDING ENVELOPE	-	10^2 μCi/cc of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.

SAMPLING MEDIA

- Iodine > 90 percent effective adsorption for all forms of gaseous iodine.
- Particulates > 90 percent effective retention for 0.3 micron (m) diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.
- Entrained moisture in effluent stream should not degrade adsorber.
- Continuous collection required whenever exhaust flow occurs.
- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
- Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

Attachment 3 Containment High-Range Radiation Monitor

Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Clarification

(1) Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-3 (of NUREG-0737, presented below).

(2) The specification of 10^8 rad/hr in the above position was based on a calculation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments, but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 R/hr.

(3) The monitors shall be located in containment(s) in a manner which will provide a reasonable assessment of area radiation conditions inside the containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

(4) For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

(5) The monitors are required to respond to gamma photons with energies as low as 60 keV and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV, as specified in Table II.F.1-3 of NUREG-0737. Monitors that use thick shielding to increase the upper range will underestimate postaccident radiation levels in containment by

several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	-	The capability to detect and measure the radiation level within the reactor containment during and following an accident.
RANGE	-	1 rad/hr to 10^8 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^7 R/hr (gamma only).
RESPONSE	-	60 keV to 3 MeV photons, with linear energy response ±20%) for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.
REDUNDANT	-	A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).
DESIGN AND QUALIFICATION	-	Category 1 instruments as described in Appendix B, except as listed below.
SPECIAL CALIBRATION		In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.
SPECIAL ENVIRONMENTAL QUALIFICATIONS	-	Calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.

Attachment 4 Containment Pressure Monitor

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel and -5 psig for all containments.

Clarification

(1) Design and qualification criteria are outlined in Appendix B (of NUREG-0737).

(2) Measurement and indication capability shall extend to 5 psia for subatmospheric containments.

(3) Two or more instruments may be used to meet requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

(4) Continuous display and recording of the containment pressure over the specified range in the control room is required.

(5) The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

Attachment 5 Containment Water Level Monitor

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

Clarification

(1) The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in Appendix B (of NUREG-0737). The narrow-range channel shall meet the requirements of Regulatory Guide 1.89.

(2) The measurement capability of 600,000 gallons is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement, based on the available water supply capability at their plant.

(3) Narrow-range water level monitors are required for all sizes of sumps but are not required in those plants that do not contain sumps inside the containment.

(4) For BWR pressure-suppression containments, the emergency core cooling system (ECCS) suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

(5) The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

Attachment 6 Containment Hydrogen Monitor

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10 percent hydrogen concentration under both positive and negative ambient pressure.

Clarification

(1) Design and qualification criteria are outlined in Appendix B (of NUREG 0737).

The continuous indication of hydrogen concentration is not required during normal operation.

(2) If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

(3) The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

18.2.12.2 Union Electric Response

Radiological Noble Gas Effluent Monitors

The Callaway design provides a wide range noble gas radiation monitor for each of the release paths listed below. Each monitor will include detectors covering the range shown below:

MONITOR	RANGE
Plant unit vent (GT-RE-21B)	10^{-7} to 10^{+5} μCi/cc
Radwaste building effluent (GH-RE-10B)	10^{-7} to 10^{+5} μCi/cc

The locations of these monitors are shown on Radiation Zone Drawing Figure 12.3-2, Sheet 4. Separate monitoring capability for the condenser air removal system is not provided because this system exhausts through the plant vent. The SNUPPS design includes gamma detectors to monitor the plume from the main steam power-operated relief valves and to monitor the steam discharge from the turbine-driven auxiliary feedwater pump. Additional information on this monitoring system is provided by Reference 10.

Continuous indication is provided in the control room for each monitor. Each monitor is recorded in the control room.

The system/methods for monitoring and analysis are described in Reference 10. The readouts from the wide range monitors are input to the plant computers. This information is accessible from the technical support center and the emergency operations facility.

The procedures used to calibrate the instruments and calculate release rates have been incorporated into the Callaway Plant procedures.

The following additional information was provided by Reference 10.

a. System description information, including energy dependence or response, range and sensitivity with respect to Xe-133, vendor model number, and methods used to assure representative measurements and background correction.

b. The calculational methods or procedures to be used for converting instrument readings to release rate per unit time based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

Union Electric submitted a variance request (to NUREG-0737, II.F.1, Attachment 1, Clarification 1) to manually correct the wide range gas monitor display to account for noble gas spectrum changes with time using the emergency plan procedures (See ULNRC-1393, dated October 24, 1986). Additional information to support this variance was submitted to NRC in ULNRC-1796, dated June 21, 1988 and ULNRC-1840, dated October 5, 1988. This variance was accepted by NRC by letter dated February 1, 1989.

Union Electric also submitted information on the noble gas radiation monitors concerning location and Technical Specifications in response to verbal NRC questions (See ULNRC-1825, dated September 2, 1988). This issue is considered closed by Union Electric. (Note: The Radiological and Environmental Technical Specifications were removed from the Technical Specifications and relocated to the Offsite Dose Calculation Manual and the Process Control Program; see ULNRC-2070, dated September 6, 1989 and Amendment 50 dated February 12, 1990, to the Callaway License.)

Provisions for Continuous Sampling of Plant Effluents for Post-Accident Releases of Radioiodines and Particulates

The design provides for continuous sampling of effluent radioiodines and particulates. The wide range gas monitors described above include the capability to obtain grab samples. The sampling is accomplished by adsorption of iodine on charcoal filters or other media and by use of particulate filters. The sampling system criteria for all airborne monitoring systems are provided in Section 11.5.2.3.1.2 of the FSAR. After collection, laboratory analyzers can be used to quantify iodine and particulate releases. A backup power source is provided for sample collection and analysis equipment to ensure operation for a minimum of 7 consecutive days. The procedures for each facility discuss the methods and counting equipment used to determine releases. The expected doses from obtaining and counting a sample have been calculated to range between 750 and 1300 mrem for a sample at the unit vent. These doses meet the requirement of NUREG-0737. Additional information regarding how the design meets the recommendations of Table II.F.1-2 and the provisions for approximate isokinetic sampling was provided by Reference 10.

A variance request to NUREG-0737, II.F.1, Attachment 2, Clarification 3 to leave the wide range gas monitor sample nozzle sized to large to produce isokinetic flow was submitted by ULNRC-1839, dated October 4, 1988. Additional information to support this variance was submitted by ULNRC-2011, dated June 2, 1989. This variance was accepted by NRC by letter dated August 4, 1989.

Union Electric submitted a variance request to NUREG-0737, II.F.1, Attachment 2, Clarification 1 concerning the wide range gas monitor. This variance (submitted in ULNRC-1095, dated May 14, 1985) stated that UE could not perform empirical determination of line loss correction factors (iodine plateout) for post accident iodine sampling from the unit vent wide range gas monitor. Additional supporting information was submitted in ULNRC-1255, dated February 10, 1986.

Containment Radiation Monitors

The Callaway design meets the recommendations of Table II.F.1-3. The design includes two physically separated Class 1E containment radiation monitors. The monitors are designated as 0-GT-RE-59 and 0-GT-RE-60. The detectors are located inside containment.

Indication is provided in the control room for each monitor, which is powered from a Class 1E power source. One channel is provided with a recorder, which is powered from Class 1E power sources. Each monitor has a range up to 10^8 R/hr for gamma radiation. The monitors are sensitive down to 60 keV photons. The response of the monitors is linear (±20%) for energies between .1 Mev and 3 Mev. The equipment has been seismically qualified for the location in which it is installed. The components are environmentally qualified for the environmental conditions to which they will be subjected.

Calibration of the monitors is addressed in procedures.

Additional information regarding the details of the design is provided in Section 11.5.2.3.2.4 and Reference 10.

Union Electric submitted a variance request to NUREG-0737, II.F.1, Attachment 3 (See ULNRC-1441, dated January 29, 1987) from the requirement to calibrate the high range radiation monitor for at least one point per decade of range between 1 R/hr and 10^3 R/hr. By letter dated October 16, 1989, the NRC rejected this variance request. Union Electric has committed to replace these detectors at Refuel IV with detectors that are calibrated to the requirements of NUREG-0737.

Containment Pressure Indication

The Callaway design provides a dual range, redundant, continuous indication of containment pressure with both ranges (0 to 69 psig and -5 to 180 psig) indicated and recorded in the control room at the same time. The extended range indicators are Class 1E; the extended range recorder is isolated from the Class 1E circuitry and is non-Class 1E. As a minimum, their range is from minus 5 psig to three times the containment design pressure of 60 psig.

The response time of the containment pressure control room indication is 10 seconds for both the narrow and wide-range instrument channels. The accuracy of both the narrow and wide range channels is ±4 percent of scale. The pressure monitor instrumentation meets the design and qualification criteria of NUREG-0737, Appendix B in accordance with the WCAP 8587 qualification reference of FSAR Table 3.11(B)-3.

Containment Water Level Indication

The Callaway design includes in the control room continuous indication of the containment water level. This instrumentation is redundant and designed and qualified in accordance with Class 1E requirements to meet the requirements of NUREG-0737, Appendix B. A single range is used to monitor both the containment normal sump level and the containment water level. The range is 13 feet, which covers 6 inches from the bottom of the containment normal sump to an elevation equivalent to 650,000 gallons. The upper limit of the range is greater than the maximum calculated water level. The accuracy of the indication is ±4 percent. This accuracy is sufficient for the purpose of verifying that adequate water level (NPSH) is available to the pumps taking suction from the containment. The switchover of the low pressure safety injection pumps to recirculation is accomplished without the use of the containment water level indication.

Containment Hydrogen Concentration Monitor

The present design includes redundant safety-grade (Class 1E) containment post-LOCA hydrogen analyzers with redundant Class 1E indication provided in the control room. These monitors meet the design and qualification requirements of NUREG-0737, Appendix B. The hydrogen analyzers have a range of 0-10 percent hydrogen volume and are designed to operate under minimum and maximum containment design pressure.

The hydrogen analyzers are manually initiated following an event. Once initiated, they provide a continuous measurement of hydrogen concentration within 30 minutes.

The sample points for the containment hydrogen monitors are in the vicinity of the intake of the containment air coolers and the post-accident water level.

18.2.12.3 Conclusion

The Callaway design provides six additional post-accident monitors specified in NUREG-0737 for accident diagnosis and mitigation. Callaway has developed emergency operating procedures which detail the use of each instrument specified during an accident. The Callaway design meets the intent of NUREG-0737 with the exception of the aforementioned variances.

The design is consistent with the recommendations of NUREG-0737, item II.F.1, for noble gas monitors.

The design includes features to sample plant effluents under accident conditions. The design of sampling system satisfies the criteria in NUREG-0737, item II.F.1.

The containment radiation monitor design meets the recommendations of item II.F.1-3.

The extended range containment pressure monitor design meets the recommendations of item II.F.1-4.

The design for containment water level indication meets the requirements of NUREG-0737, item II.F.1-5.

The design for the containment hydrogen monitors meet the requirements of NUREG-0737, item II.F.1-6.

18.2.13 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

18.2.13.1 NRC Guidance Per NUREG-0737

Position

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

Clarification

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

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

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

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

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

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

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

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

(7) All instrumentation in the final ICC system must be evaluated for conformance to Appendix B (to NUREG-0737), "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

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

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

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

- (a) The use of this information by an operator during both normal and abnormal plant conditions.
- (b) Integration into emergency procedures.
- (c) Integration into operator training.

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

ATTACHMENT 1, DESIGN AND QUALIFICATION CRITERIA FOR PRESSURIZED WATER REACTOR INCORE THERMOCOUPLES

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

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

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

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

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

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

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

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

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

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

(5) The instrumentation must be evaluated for conformance to Appendix B (to NUREG-0737), "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of items 6 through 9 which follow.

(6) The primary and backup display channels should be electrically independent, energized from independent station Class 1E power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. The primary display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source, battery backed, where momentary interruption is not tolerable. The backup display and associated hardware should be Class 1E.

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

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

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

18.2.13.2 Union Electric Response

Item II.F.2 of NUREG-0737 specifies the following as required documentation concerning instrumentation for detection of inadequate core cooling (ICC):

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

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

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

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

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

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

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

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

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

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

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

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

The following is a discussion of each of the above items as they relate to the Callaway instrumentation for detection of ICC:

(1) The final system to be used at Callaway to detect ICC consists of a reactor vessel level instrumentation system and a thermocouple/core cooling monitor system.

Reactor Vessel Level Instrumentation System (RVLIS)

The design provides redundant safety-grade (Class 1E) reactor vessel water level instrumentation. The four reactor vessel water level indicators (LI-1311, LI-1312, LI-1321, and LI-1322) are located on the main control board reactor auxiliaries console, RL-021. The RVLIS (Figure 18.2-13) utilizes two sets of two d/p cells. These cells measure the pressure differential between the bottom of the reactor vessel and the top of the vessel. This d/p measuring system utilizes cells of differing ranges to cover different flow behavior with and without pump operation as discussed below:

(a) Reactor Vessel - Narrow Range (ΔP_b)

This measurement provides an indication of reactor vessel level from the bottom of the reactor vessel to the top of the reactor during natural circulation conditions.

(b) Reactor Vessel - Wide Range (ΔP_c)

This instrument provides an indication of reactor core and internals pressure drop for any combination of operating RCPs. Comparison of the measured pressure drop with the normal, single-phase pressure drop provides an approximate indication of the relative void content or density of the circulating fluid. The indication of coolant density is significant only when the subcooling is near zero. This instrument monitors coolant conditions on a continuing basis during forced flow conditions.

To provide the required accuracy for level measurement, temperature measurements of the impulse lines are provided. These measurements, together with existing reactor coolant temperature measurements and wide-range RCS pressure, are employed to compensate the d/p transmitter outputs for differences in system density and reference leg density, particularly during the change in the environment inside the containment structure following an accident.

Additional information (i.e., analyses, evaluations) concerning the Westinghouse generic reactor vessel level instrumentation system has been submitted to the NRC via Reference 7. The specific hardware for Callaway is not exactly as documented in Reference 7, since the Callaway design does not include a separate measurement of reactor vessel level

above the hot legs (reactor vessel upper range). However, the analyses, evaluations, and conclusions contained in Reference 7 are applicable to Callaway, since they are not sensitive to the above-mentioned design difference.

Thermocouple/Core Cooling Monitor System (T/CCMS)

The T/CCMS is a core exit thermocouple/core cooling detection system which provides presentation and display of the status of the core heat removal capability to both the plant operators and the technical support center. In the control room, the two subcooling temperature indicators are located immediately above the four level indicators on the vertical portion of the control board, RL-022. The core exit thermocouple display is mounted on the subcooling monitor cabinet, RP-081. The system consists of redundant channels and output trains of thermocouple measurements, wide-range hot- and cold-leg RTD temperatures, and reactor pressure signals. These parameters are used by the system to display thermocouple temperatures and to calculate saturation temperatures and margin of saturation (Tsat margin), which is often referred to as subcooling. The calculations are performed by the system which is based on microprocessor and data handling devices.

Thermocouple Monitor

The core exit thermocouple portion of the ICC system is arranged as follows:

(a) Primary system

The primary system measures all the thermocouples via isolators located in the qualified backup system cabinet.

(b) Backup system

The backup system consists of two channels, each monitoring half of the 50 core outlet thermocouples.

The system has separation and redundancy as well as qualification to comply with Appendix B of NUREG-0737 (see the discussion of item (4) below).

Core Cooling Monitor

The core cooling monitor portion of ICC system compares core outlet thermocouple temperatures and hot- and cold-leg RTD temperatures with the saturation temperature based on the lowest of three pressure signals.

This system has separation and redundancy as well as qualification to comply with Appendix B of NUREG-0737 (see the discussion of item (4) below).

One of the indicators of an approach to an ICC situation is the response of the core exit thermocouples (T/Cs) to the presence of superheated steam. The core exit thermocouples do not provide an indication of the amount of core voiding. Response of the core exit T/Cs provides a direct indication of the existence of ICC, the effectiveness of ICC recovery actions, and restoration of adequate core cooling. The core is adequately cooled whenever the vessel mixture level is above the top of the core, and the core may have a significant void fraction and still be adequately cooled.

The thermocouple/core cooling monitor combines the functions of monitoring for excessive core exit thermocouple temperatures and monitoring both core exit thermocouple temperatures and hot- and cold-leg RTD temperatures for saturation margin (Tsat meter).

The system consists of two redundant channels, each monitoring half of the core outlet thermocouples, and four hot- and cold-leg RTDs. Three reactor pressure input signals are used with the auctioneered low pressure used by the microprocessor to perform the Tsat margin function. The thermocouple temperatures are corrected for reference junction temperature with three reference junction temperature signals input to each channel. (All of the thermocouples connected to one channel are from one reference junction unit.)

The system's two redundant trains utilize the following safety-grade equipment:

(a) Thermocouples

(b) Reference junction boxes

(c) RTDs

(d) Termination board assemblies

(e) Microprocessor assemblies

(f) Remote displays

(g) Analog meters

(h) Recorders

(i) Power supplies

(j) Connections and cabling

The equipment listed above and shown in Figure 18.2-12 has been designed to satisfy the requirements of IEEE Standard 279. This safety-grade system is isolated from the non-Class 1E plant computer, technical support center, and data links by qualified isolation devices. Details of isolation device qualification were provided in SLNRC 84-100, dated June 29, 1984 and SLNRC 84-104, dated July 26, 1984.

The system can display individual thermocouple temperatures and provides two levels of alarm when preset temperatures are exceeded. The display is an alphanumeric panel digital display with 8 lines of 32 characters each located at the processing cabinets, behind the main control board.

The thermocouple monitor can calculate and display core outlet temperature quadrant tilts based on thermocouple temperatures. The tilts calculated by each unit are based on half the total number of core thermocouples. This information is also available to the operator at the main control board via the plant computer.

The core cooling monitor compares core outlet thermocouple temperatures and hot- and cold-leg RTD temperatures with the saturation temperature based on the lowest of three pressure signals. Two levels of alarm are provided for the core cooling (Tsat) monitor function. The margin to saturation is displayed on two redundant analog meters on the vertical section of the main control board and are visible to an operator at the control console.

The thermocouple/core cooling monitor provides information to the operator that assists in the performance of the required manual safety functions following a Condition II, III, or IV event. This includes information relative to maintaining the plant in a safe shutdown condition or to proceeding to a cold shutdown condition consistent with the Technical Specification limits.

At Callaway, the core exit T/Cs protrude slightly from the bottom of the upper core plate support columns. In this location, they measure the temperature of the fluid leaving the core region through the flow passages in the upper core plate. Flow from the upper head must enter the upper plenum via the control rod drive guide tubes before being able to enter the upper core plate flow passages. In addition, the LOCA blowdown depressurization behavior must be such that there is a flow reversal for the

core exit T/Cs to detect the upper head fluid temperature. The upper head fluid is expected to mix with the upper plenum fluid as it drains from the upper head.

The potential for core exit T/C cooling from colder upper head fluid, while the core has an appreciable void fraction, is not viewed as a potential problem for the detection of an inadequate core cooling situation. Although some Semiscale tests indicated core voiding while the upper head was liquid solid, these tests do not imply that the core exit T/Cs would give an ambiguous indication of ICC calculations for a Westinghouse PWR, and consideration of the core exit T/C design would not result in ambiguous ICC indications.

Additional information concerning the thermocouple/ core cooling monitor system is provided in Table 18.2-3.

(2) Reference 7 provides a design analysis and evaluation of the instrumentation for detection of inadequate core cooling (ICC).

By letter dated February 20, 1986, (Reference 15) Westinghouse reported to the NRC the results of an evaluation relative to the consequences of larger than expected post-accident errors on the T/CCMS. These errors, which are identified in WCAP 8587 (Reference 16), affected subcooling margin calculations, the use of T/Cs for ICC indication and the use of T/Cs as temperature compensation for RVLIS. Based on the revised error values, restrictions were placed on use of T/Cs for RVLIS and subcooling margin calculations and the EOP Guidelines were revised to incorporate new ICC indication setpoints and revised accuracy requirements were developed for RVLIS. Modifications to plant hardware and procedures, required by the larger than expected T/CCMS errors, have been implemented at Callaway.

(3) Additional testing of the equipment described above has been completed in order to establish and upgrade qualification of the equipment to comply with NUREG-0737.

The test programs were:

(a) Qualification tests of core exit thermocouples

(b) Qualification tests of reference (temperature compensation) junction boxes

(c) Qualification tests of electronics to add to the system computer and technical support center isolators and microprocessors

(d) Qualification of isolation devices, cables and connectors, reference leg RTDs and hydraulic isolators.

(4) An evaluation of the conformance of the reactor vessel level instrumentation system to NUREG-0737 is provided in Reference 7.

An evaluation of the conformance of the thermocouple/ core cooling monitor system to NUREG-0737 (Attachment 1 and Appendix B) is as follows:

(a) Attachment 1, Item (1)

The core exit thermocouples have been qualified so as to comply with the recommendations of Regulatory Guides 1.89 and 1.100, as clarified hereinafter. The thermocouples are located at the core exit and in an arrangement such that each of the redundant microprocessor systems has core exit temperatures distributed over the entire core, in sufficient number to determine the radial power distribution and so located as to verify power distribution symmetry among core quadrants.

(b) Attachment 1, Item (2)

The primary operator display is a computer-based display and calculation system. It provides information as required by subitems (a) through (f) of Attachment 1, Item (2), in Section 18.2.13.1.

(c) Attachment 1, Item (3)

The backup system to display thermocouple readings is located in a cabinet which also houses the core cooling monitor. Backup system display is accomplished by the Class 1E ICC instrumentation including the microprocessor assemblies, remote displays, analog meters, and recorders. This system has a 40-character (alphanumeric) display line located on the front of each of the microprocessor drawers. This backup system can display all of the 50 individual thermocouple temperatures within 6 minutes. The range extends from 0°F to 2500°F.

(d) Attachment 1, Item (4)

Human factors consideration of the types and locations of displays and alarms is discussed in Sections 18.1.16 and 18.3.2. The ICC instrumentation has been considered in the overall human factors evaluation.

(e) Attachment 1, Item (5)

Conformance to the specific items of Appendix B to NUREG-0737 is as follows:

1) Appendix B, Item (1)

The thermocouple/core cooling monitor instrumentation has been tested to establish environmental qualification in accordance with Regulatory Guide 1.89 (NUREG-0588). This qualification requirement applies to the complete instrumentation channel from thermocouple to display where display indicates the remote display, analog meter, and recorder. Qualified channel isolation devices isolate this qualified instrumentation from the data links, technical support center display, and plant computer display.

The seismic portion of the environmental qualification testing has been performed to comply with Regulatory Guide 1.100. This seismic qualification provides assurance that the instrumentation will continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake.

Instrumentation whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable have been qualified using the following criteria.

The qualification environment is based on the design basis accident events, except the assumed maximum of the value of the monitored variable is the value equal to the maximum range for the variable. The monitored variable is assumed to approach this peak by extrapolating the most severe initial ramp associated with the design basis accident events. The decay for this variable is considered proportional to the decay for this variable associated with the design basis accident events. No additional qualification margin needs to be added to the extended range variable. All environmental envelopes except that pertaining to the variable measured by the information display channel are those associated with the design basis accident events.

The above environmental qualification requirement does not account for steady-state elevated levels that may occur in other environmental parameters associated with the

extended range variables. For example, a sensor measuring containment pressure must be qualified for the measured process variable range, but the corresponding ambient temperature is not mechanistically linked to that pressure. Rather, the ambient temperature value is the bounding value for design basis accident events analyzed in Chapter 15.0. The extended range requirement ensures that the equipment will continue to provide information should conditions degrade beyond those postulated in the safety analysis. Since variable ranges are nonmechanistically determined, extension of associated parameter levels is not justifiable and has, therefore, not been required.

2) Appendix B, Item (2)

The purpose for qualifying the thermocouple/ core cooling monitoring system is to generate evidence that the equipment will maintain and perform its functions during a design basis event. It is of special concern during the qualification effort to uncover common mode failures.

The single-failure criteria for the computer and information beyond the isolator does not apply to this data-based information device as referred to in NUREG-0737 (clarification item (8)). In relation to diversification, the use of reactor vessel level instrumentation adds diversification to the ICC instrumentation. Inclusion of the core cooling (Tsat margin) monitoring functions enhances even further the capability of the Callaway ICC instrumentation.

3) Appendix B, Item (3)

The instrumentation is energized from Class 1E power sources.

4) Appendix B, Item (4)

Although not specifically recommended by NUREG-0737, the ICC instrumentation complies with the applicable portions of IEEE Standard 279. The systems utilize two trains; therefore, the "Exemption" as defined in Paragraph 4.11 of IEEE Standard 279 is applicable here.

5) Appendix B, Item (5)

The ICC equipment falls under the quality assurance requirements applicable to Class 1E equipment. Refer to Appendix 3A for a discussion of the quality assurance regulatory guides.

6) Appendix B, Item (6)

A digital (40-character) line display is provided for the thermocouple readings in the backup system. The computer-based (primary) thermocouple indication system has continuous (recording) displays.

7) Appendix B, Item (7)

The backup Class 1E system (which is on demand) includes redundant recorders. The computer-based (primary) indication system has continuous (recording) displays.

8) Appendix B, Item (8)

The instruments are specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

9) Appendix B, Item (9)

The Callaway ICC instrumentation complies with isolation requirements.

10) Appendix B, Item (10

The Callaway ICC instrumentation is testable as required.

11) Appendix B, Item (11)

Servicing, testing, and calibrating programs are specified to maintain the capability of the monitoring instrumentation.

12) Appendix B, Item (12)

The access to the thermocouple/core cooling monitor permits removing channels for service (location is a main control room). The testing and/or maintenance is facilitated by this system location.

13) Appendix B, Item (13)

The design facilitates administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

14) Appendix B, Item (14)

The monitoring instrumentation design minimizes the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator.

15) Appendix B, Item (15)

The instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

16) Appendix B, Item (16)

The instrumentation used in both the reactor vessel level instrumentation system and thermocouple monitoring receives input signals directly from the sensors that measure the parameters. The core cooling monitor also derives most of the signals directly from the sensors except in the case where Tsat pressures and others are obtained from the protection set.

17) Appendix B, Item (17)

The instruments used for ICC instrumentation are also used, with the exception of the reactor vessel level indication, for monitoring normal operation of the plant to the extent that it is practical. No loss of sensitivity is expected due to this use.

18) Appendix B, Item (18)

Periodic testing is in accordance with the applicable portions of Regulatory Guide 1.118.

(f) Attachment 1, Item (6)

The instrumentation system power supplies are in conformance with this requirement. However, the required circuits to the thermocouples are separated only to the maximum extent possible.

(g) Attachment 1, Item (7)

The instrumentation qualification is discussed in item (4).e.1 above.

(h) Attachment 1, Item (8)

The instrumentation system is in conformance with this requirement.

(i) Attachment 1, Item (9)

Quality assurance is discussed in item (4).e.5 above.

(5) The nonredundant plant computer displays the thermocouple and Tsat functions for the primary display. These functions are performed by the Class 1E ICC instrumentation. The Class 1E microprocessor performs the calculations and provides the signals to both the primary and Class 1E backup display.

(6) In general, the system electronics are verified, maintained, and calibrated on-line by placing one of the redundant trains into a test and calibrate mode while leaving the other train in operation to monitor inadequate core cooling.

A general verification was performed before shipment, but plant specific data was not used. The capability exists for the operator to verify the operation of the system. This involves disconnecting the sensors at the RVLIS electronics, providing an artificial input, and observing the response of the system on the front panel and remote display.

The "7300" RVLIS incorporates circuit cards that provide an output proportional to the change in resistance of the RTD. The card contains a resistance bridge driven by a power supply to produce a signal proportional to the changes in resistance of the RTD, and a signal characterizer which accommodates linear calibration of non-linear RTDs.

On-line calibration of the system is made possible by the "card edge" adjustments. The circuit cards were calibrated at the factory; however, if the function is changed or a component on the card is replaced, the calibration procedure is given within the equipment reference manual.

(7) and (8) The Westinghouse Owners Group has developed ICC operating guidelines. These guidelines were developed using the generic ICC analyses discussed in Section 18.1.8. These generic guidelines were considered, as appropriate, by Callaway in developing plant specific operating procedures.

(9) No additional submittals are required, with the exception of emergency operating procedures.

18.2.13.3 Conclusion

The Callaway instrumentation for detection of ICC meets the intent of guidance in NUREG-0737, Item II.F.2.

18.2.14 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT (II.G.1)

18.2.14.1 NRC Guidance Per NUREG-0737

Position

Consistent with satisfying the requirements for General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Clarification

(1) Although the primary concern resulting from lessons learned from the accident at TMI is that the PORV block valves must be closable, the design should retain, to the extent practical, the capability to also open these valves.

(2) The motive and control power for the block-valve should be supplied from an emergency power bus different from the source supplying the PORV.

(3) Any changeover of the PORV and block-valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

(4) For those designs in which instrument air is needed for operation, the electrical power supply should be required to have the capability to be manually connected to the emergency power sources.

18.2.14.2 Union Electric Response

The pressurizer level indication channels are powered from vital, Class 1E buses and displayed in the control room. These buses are described in Section 8.3; they are capable of being supplied from onsite emergency power (diesel generators) or offsite power.

The pressurizer PORVs and block valves are powered from vital, Class 1E power sources. The separation group assignment is indicated on system drawings in FSAR Section 5.1.

The pressurizer PORVs are relied on to perform the following safety functions:

a. Pressure control during a shutdown concurrent with loss of offsite power;

b. Overpressure protection at low reactor coolant system pressures; and

c. RCS depressurization in the mitigation of the accidents discussed in Sections 15.5.1 and 15.6.3.

These functions are described in Sections 5.2 and 5.4 (A).

The PORV block valve is provided to isolate the PORV should the PORV develop an unacceptable leakage during operation.

The pressurizer level indication is used during normal operation to control pressurizer level (see Figure 7.2-1, sheet 11).

The pressurizer level indication is used for the reactor trip logic and is a displayed parameter for safe shutdown control. The safety design basis of the pressurizer level indication is provided in Section 7.2 and Section 7.5.

18.2.14.3 Conclusion

The Callaway design for the emergency power for pressurizer equipment satisfies Item II.G.1 of NUREG-0737. The Callaway design proposes an alternative to the power supply assignment proposed for the pressurizer PORVs and PORV block valves. The alternative is justified based on the diversity in power supply assignments for these

valves, i.e., motor-operated (AC) block valves and solenoid-operated (DC) PORVs and based on the above requirements for PORV use.

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

Position

"(C.1.5) - Review all valve positions, positioning requirements, positive controls, and related tests and maintenance procedures to ensure proper ESF functioning. See Bulletins 79-06A Item 8, 79-06B Item 7, and 79-08 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 Bulletin 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 SNUPPS facilities were carried out as a joint effort between Union Electric, Wolf Creek Nuclear Operating Corporation, and other consultants. This development and review effort has considered the concerns of Items C.1.5 and C.1.10 of NUREG-0694 and Union Electric has performed the actions required by the applicable I&E Bulletin sections.

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

18.2.15.3 Conclusion

Union Electric has developed and reviewed plant procedures in accordance with the NRC guidance in II.K.1 of NUREG-0694.

18.2.16 ORDERS ON FACILITIES WITH BABCOCK & WILCOX NUCLEAR STEAM SUPPLIER SYSTEMS (II.K.2)

18.2.16.1 Control of Auxiliary Feedwater Independent of the Integrated Control System (II.K.2.2)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.2 Auxiliary Feedwater System Upgrading (II.K.2.8)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.3 Failure Mode Effects Analysis on the Integrated Control System (II.K.2.9)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.4 Safety-Grade Anticipatory Reactor Trip (II.K.2.10)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.5 Thermal Mechanical Report--Effect of High-Pres- sure Injection on Vessel Integrity for Small-Break

Loss-of-Coolant Accident with no Auxiliary Feedwater (II.K.2.13)

18.2.16.5.1 NRC Guidance Per NUREG-0737

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Clarification

The position deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow. One aspect that bears heavily on the effects of safety injection flow is the mixing of safety injection water with reactor coolant in the reactor vessel. B&W provided a report on July 30, 1980 that discussed the mixing question and the basis for a conservative analysis of the potential for thermal shock to the reactor vessel. Other PWR vendors are also required to address this issue with regard to recovery from small breaks with an extended loss of all feedwater. In particular, demonstration shall be provided that sufficient mixing would occur of the cold high-pressure injection (HPI) water with reactor coolant so that significant thermal shock effects to the vessel are precluded.

18.2.16.5.2 Union Electric Response

Westinghouse (in support of the Westinghouse Owners Group) has developed a method and has performed analyses for a spectrum of small loss-of-coolant accidents. The method employs the NOTRUMP computer program to generate the thermal/ hydraulic transients. The thermal transients on the reactor vessel beltline and the inlet nozzle are analyzed based on the thermal/hydraulic data from the NOTRUMP code. The Westinghouse developed pressurized thermal shock evaluation (PTS) methodology has been submitted to and approved by the NRC (Reference 13). The Callaway calculated RT_{NDT} values are well below the screening criterion of 10 CFR 50.61.

(See ULNRC-1244, dated January 21, 1986; NRC letter dated December 15, 1986; ULNRC-1675, dated November 10, 1987 and ULNRC-1867, dated November 30, 1988.)

18.2.16.6 Effects of Slug Flow on Steam Generator Tubes (II.K.2.15)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.7 Reactor Coolant Pump Seal Damage (II.K.2.16)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.8 Potential for Voiding in the Reactor Coolant System During Transients (II.K.2.17)

18.2.16.8.1 NRC Guidance Per NUREG-0737

Position

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

Clarification

The background for this concern and a request for this analysis was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from R. W. Reid, NRC, to all B&W operating plants, dated January 9, 1980.

18.2.16.8.2 Union Electric Response

Westinghouse (in support of the Westinghouse Owners Group) has performed a study which addresses the potential for void formation in Westinghouse-designed nuclear steam supply systems during natural circulation cooldown/depressurization transients.

This study has been submitted to the NRC by the Westinghouse Owners Group (Ref. 1) and is applicable to the Callaway Plant.

In addition, the Westinghouse Owners Group has developed appropriate modifications to the Westinghouse Owners Group Emergency Response Guidelines (ERGs) to take the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and to specify those conditions under which upper head voiding may occur. The SNUPPS utilities have considered the generic guidance developed by the Westinghouse Owners Group in the development of plant specific operating procedures.

18.2.16.9 Sequential Auxiliary Feedwater Flow Analysis (II.K.2.19)

Not applicable to Westinghouse pressurized water reactors.

18.2.16.10 Small-Break Loss-of-Coolant Accident Which Repressurizes the Reactor Coolant System to the Power-Operated Relief Valve Set Point (II.K.2.20)

Not applicable to Westinghouse pressurized water reactors.

18.2.17 RECOMMENDATIONS FROM THE BULLETINS AND ORDERS TASK FORCE (II.K.3)

18.2.17.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (II.K.3.1)

18.2.17.1.1 NRC Guidance Per NUREG-0737

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to ensure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Clarification

Implementation of this action item was modified in the May 1980 version of NUREG-0660. The change delays implementation of this action item until after the studies specified in TMI Action Plan item II.K.3.2 have been completed, if such studies confirm that the subject system is necessary.

18.2.17.1.2 Union Electric Response

Westinghouse, as a part of the response prepared for the Westinghouse Owners Group to address item II.K.3.2 (refer to Section 18.2.17.2), has evaluated the necessity of incorporating an automatic pressurizer power-operated relief valve isolation system. This evaluation is documented in Reference 2 and concluded that such a system should not be required. The Callaway design includes the capability to remote-manually isolate the power-operated relief valves by closing block valves from the main control room.

18.2.17.1.3 Conclusion

Based on the above discussion, Callaway meets the intent of the guidelines of NUREG-0737, Item II.K.3.1.

18.2.17.2 <u>Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (II.K.3.2)</u>

18.2.17.2.1 NRC Guidance Per NUREG-0737

Position

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open, power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Clarification

Based on its review of feedwater transients and small LOCAs for operating plants, the Bulletins and Orders Task Force in the Office of Nuclear Reactor Regulation recommended that a report be prepared and submitted for staff review which documents the various actions that have been taken to reduce the probability of a small-break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements in reactor safety. Action Item II.K.3.2 of NUREG-0660, published in May 1980, changed the implementation of this recommendation as follows: In addition to modifications already implemented on PORVs, the report specified above should include safety examination of an automatic PORV isolation system identified in Task Action Plan item II.K.3.1.

Modifications to reduce the likelihood of a stuck-open PORV will be considered sufficient improvements in reactor safety if they reduce the probability of a small-break LOCA

caused by a stuck-open PORV such that it is not a significant contributor to the probability of a small-break LOCA due to all causes. (According to WASH-1400, the median probability of a small-break LOCA S_2 with a break diameter between 0.5 inches and 2.0 inches is 10^{-3} per reactor-year with a variation ranging from 10^{-2} to 10^{-4} per reactor-year.)

The above-specified report should also include an analysis of safety-valve failures based on the operating experience of the pressurized-water-reactor (PWR) vendor designs. The licensee has the option of preparing and submitting either a plant-specific or a generic report. If a generic report is submitted, each licensee should document the applicability of the generic report to his own plant.

Based on the above guidance and clarification, each licensee should perform an analysis of the probability of a small-break LOCA caused by a stuck-open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. This analysis shall evaluate the effect of an automatic PORV isolation system specified in Task Action Plan, Item II.K.3.1. In evaluating the automatic PORV isolation system, the potential of causing a subsequent stuck-open safety valve and the overall effect on safety (e.g., effect on other accidents) should be examined.

Actual operational data may be used in this analysis, where appropriate. The bases for any assumptions used should be clearly stated and justified.

The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve a sufficient amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Task Action item II.K.3.1 is necessary.

In addition to the analysis described above, the licensee should compile operational data regarding pressurizer safety valves for PWR vendor designs. These data should then be used to determine safety-valve failure rates.

The analyses should be documented in a report. If this requirement is implemented on a generic basis, each licensee should review the appropriate generic report and document its applicability to his own plant(s). The report and the documentation of applicability (where appropriate) should be submitted for NRC staff review by the specified date.

18.2.17.2.2 Union Electric Response

As mentioned in item II.K.3.1 above (Section 18.2.17.1), the Westinghouse Owners Group has submitted a Westinghouse-prepared report (Ref. 2) which provides a probabilistic analysis to determine the probability of a PORV LOCA, estimates the effect of the post-TMI modifications, evaluates an automatic PORV isolation concept and concludes that an automatic isolation capability is not required and provides PORV and

safety valve operational data for Westinghouse plants. Because of the sensitivity analyses included in the report, the report is generic and is applicable to Callaway. The report identifies a significant reduction in the PORV LOCA probability as a result of post-TMI modifications, and the calculations compare favorably with the operational data for Westinghouse plants (included as an appendix to the report).

18.2.17.2.3 Conclusion

The requirements of this item were resolved by submittal of the analysis report discussed in Reference 2.

18.2.17.3 Reporting Safety and Relief Valve Failures and Challenges (II.K.3.3)

18.2.17.3.1 NRC Guidance Per NUREG-0694

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

18.2.17.3.2 Union Electric Response

The failure of a PORV to close on demand and a failure of a primary system safety valve to close will no longer be reported per Generic Letter 97-02.

18.2.17.3.3 Conclusion

Union Electric's commitment documented above meets the requirements of NUREG-0737, II.K.3.3.

18.2.17.4 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident (II.K.3.5)

18.2.17.4.1 NRC Guidance Per NUREG-0737

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in the safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Clarification

This action item has been revised in the May 1980 version of NUREG-0660 to provide for continued study of criteria for early reactor coolant pump trip. Implementation, if any is required, will be delayed accordingly. As part of the continued study, all holders of approved emergency core cooling (ECC) models have been required to analyze the forthcoming LOFT test (L3-6). The capability of the industry models to correctly predict the experimental behavior of this test will have a strong input on the staff's determination of when and how the reactor coolant pumps should be tripped.

18.2.17.4.2 Union Electric Response

In response to IE Bulletin No. 79-06C, Westinghouse (in support of the Westinghouse Owners Group) performed an analysis of delayed reactor coolant pump (RCP) trip during small-break LOCAs. This analysis is documented in Reference 3 and is the basis for the Westinghouse and SNUPPS position on RCP trip (i.e., automatic RCP trip is not necessary since sufficient time is available for manual tripping of the RCPs).

Westinghouse (again in support of the Westinghouse Owners Group) has performed test predictions of the LOFT Experiment L3-6. The results of these predictions are documented in References 4 and 5. The results constitute both a best estimate model prediction with the NOTRUMP computer program and an evaluation model prediction with the WFLASH computer program, using the supplied set of initial boundary assumptions.

By letter dated February 8, 1983, the NRC issued Generic Letter 83-10c. The NRC concluded that each nuclear plant applicant should determine the need to trip the reactor coolant pumps following an accident or transient. By SNUPPS letters dated April 22, 1983 and April 13, 1984, (SLNRC 83-21 and SLNRC 84-66) responses to Generic Letter 83-10c were provided. The responses referenced previous Westinghouse Owner's Group reports dated December 1, 1983 and March 9, 1984.

The NRC has issued Generic Letter 85-12 which confirmed the acceptability of the information provided by the Westinghouse Owner's Group, in response to Generic Letter 83-10, and requested that plant-specific information be submitted to the NRC. By letter dated November 27, 1985 (ULNRC-1215), UE responded to Generic Letter 85-12. By letter dated August 4, 1988, the NRC staff found that Union Electric addressed the Generic Letter 85-12 criteria, that the material submitted was acceptable, and that the requirements in regard to TMI Action Item II.K.3.5 were satisfied.

18.2.17.5 <u>Evaluation of PORV Opening Probability During Overpressure Transient (II.K.3.7)</u>

Not applicable to Westinghouse pressurized water reactors.

18.2.17.6 Proportional Integral Derivative Controller Modification (II.K.3.9)

18.2.17.6.1 NRC Guidance Per NUREG-0737

Position

The Westinghouse-recommended modification to the proportional integral derivative (PID) controller should be implemented by affected licensees.

Clarification

The Westinghouse-recommended modification is to raise the interlock bistable trip setting to preclude derivative action from opening the power-operated relief valve (PORV). Some plants have proposed changing the derivative action setting to zero, thereby eliminating it from consideration. Either modification is acceptable to the staff. This represents a newly available option.

18.2.17.6.2 Union Electric Response

The Callaway design originally included a pressure integral derivative (PID) controller in the power-operated relief valve control circuit. The time derivative constant in the PID controller for the pressurizer PORV was turned to "OFF" prior to commercial operation. The appropriate plant procedure for calibrating the set points in this system reflected this decision.

Setting the derivative time constant to "OFF," in effect, removed the derivative action from the controller. Removal of the derivative action decreased the likelihood of opening the pressurizer PORV since the actuation signal for the valve was then no longer sensitive to the rate of change of pressurizer pressure. This PID controller was used in the actuation circuitry for BB-PCV-0455A. A subsequent plant modification revised the PORV actuation circuitry such that the PID controller is no longer used in the actuation circuitry for BB-PCV-0455A. The current design is reflected in Figures 7.7-4, 7.2-1 (sheets 6 and 11), and 7.6-4 (sheets 1 and 2)..

18.2.17.6.3 Conclusion

The NUREG-0737 provisions for the PID controller no longer apply to Callaway.

18.2.17.7 Proposed Anticipatory Trip Modification (II.K.3.10)

18.2.17.7.1 NRC Guidance Per NUREG-0737

Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant

basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification.

Clarification

This evaluation is required for only those licensees/applicants who propose the modification.

18.2.17.7.2 Union Electric Response

This anticipatory trip modification is included in the SNUPPS design.

The NRC has raised the question of whether the pressurizer power-operated relief valves would be actuated for a turbine trip without reactor trip below a power level of 50 percent (P-9 set point). An analysis has been performed using realistic yet conservative values for the core physics parameters (primarily reactivity feedback coefficients and control rod worths), and a conservatively high initial power, average reactor temperature (T_{AVG}), and pressurizer pressure level to account for instrument inaccuracies.

The transient was initiated from the set point for the P-9 interlock, namely 50 percent of the reactor full power level plus 2 percent for power measurement uncertainty. This is a conservative starting point, and would bracket all transients initiated from a lower power level. The core physics parameters used were the ones that would result in the most positive reactivity feedbacks (i.e., highest power levels). The steam dump valves were assumed to be actuated by the load rejection controller.

Based upon the results from the analysis, the peak pressure reached in the pressurizer would be 2,302 psia. The set point for the actuation of the pressurizer power-operated relief valves is 2,350 psia. Even including the ±20 psi pressure measurement uncertainty, there is still a margin of 28 psi between the peak pressure reached and the minimum activation pressure for the pressurizer power-operated relief valves.

An additional analysis has been performed to determine the consequences (specifically the likelihood of the pressurizer power-operated relief valves opening) of having a turbine trip due to a loss of condenser vacuum.

The major difference between this analysis and the one presented above is that now the normal steam dump system is unavailable, and the steam relief must be carried out through the atmospheric relief valves. Since there is a longer delay time before the atmospheric reliefs reach their set point (in comparison to the normal steam dump system) and their capacity is about one-half of the steam dump system, there is an increased likelihood that the pressurizer PORVs will open.

Figure 18.2-14 shows the plant operating ranges for which the pressurizer PORVs will open for a turbine trip due to a loss of condenser signal. Above 50 percent power, a turbine trip will cause a reactor trip (due to P-9 set point), and the pressurizer PORV set point will not be reached. Below a power level of 35 to 40 percent (depending on fuel burnup), the pressurizer spray rate is adequate to maintain the pressurizer pressure below the set point. Therefore, only in the narrow band between about 35 and 50 percent power will the pressurizer PORVs open for a loss of condenser.

Based upon the operating history of current plants, the chances of getting a condenser unavailable signal (and hence a turbine trip) is about 156 out of 10^7 operating hours. Assuming 98 percent plant availability and a 40-year plant lifetime, this works out to about four condenser unavailable turbine trips occurring during the normal life of a plant. Assuming an equal chance of having the plant operate anywhere between 0 and 100 percent power (an unrealistic value, since they usually operate either at a full or no load level), the chances of having a condenser unavailable signal generate a transient which would result in the opening of the pressurizer PORVs is less than one per plant lifetime.

18.2.17.7.3 Conclusion

The analysis described above demonstrates an acceptably low probability of a small LOCA caused by a stuck open PORV.

18.2.17.8 Justification use of Certain PORVs (II.K.3.11)

18.2.17.8.1 NRC Guidance Per NUREG-0694

Position

Demonstrate that the PORV installed in the plant has a failure rate equivalent to or less than the valves for which there is an operating history.

18.2.17.8.2 Union Electric Response

The PORVs to be used in the Callaway design are pilot-operated relief valves. These valves are a new design and were supplied by Crosby. The valve design was tested in the Electric Power Research Institute (EPRI) valve test program (refer to NUREG-0737, Item II.D.1 for more information). The performance of the Crosby PORVs was comparable to other designs tested. In addition, the analysis of PORVs in accordance with NUREG-0737, Item III.K.3.2 (Section 18.2.17.2) addresses valve failure rates.

18.2.17.8.3 Conclusion

Based on the EPRI testing and PORV analysis identified above, failure rates for the Callaway PORV design are adequately addressed.

18.2.17.9 Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (II.K.3.12)

18.2.17.9.1 NRC Guidance Per NUREG-0737

Position

Licensees with Westinghouse-designed operating plants should confirm that their plants have an anticipatory reactor trip upon turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for the installation of this trip.

18.2.17.9.2 Union Electric Response

The Callaway design includes an anticipatory reactor trip upon turbine trip (refer to Figure 7.2-1).

18.2.17.10 Separation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels--Analysis and Implementation II.K.3.13)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.11 Isolation of Isolation Condensers on High Radiation (II.K.3.14)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.12 Modify Break-Detection Logic to Prevent Spurious Isolation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling (II.K.3.15)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.13 Reduction of Challenges and Failures of Relief Valves--Feasibility Study and System Modification (II.K.3.16)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.14 Report on Outages of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes (II.K.3.17)

18.2.17.14.1 NRC Guidance Per NUREG-0737

Position

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one

diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

18.2.17.14.2 Union Electric Response

Union Electric will provide safety system outage information as required by regulations and the Callaway Technical Specifications.

In addition, records are retained of the maintenance, inspections, and surveillance tests of the principal items related to nuclear safety. These records can be reviewed by the NRC for additional specific data on component availability. The documentation will include: 1) outage dates and duration, 2) cause of the outage, 3) systems or components involved in the outage, and 4) corrective action taken.

18.2.17.14.3 Conclusion

By letter dated May 9, 1989, NRC removed the requirement for a five year report of ECCS outages. Union Electric reports safety system outages as required by 10 CFR 50.72, 10 CFR 50.73, the Technical Specifications and other applicable regulations. This

reporting ensures that the data requested by Item II.K.3.17 of NUREG-0737 is available to NRC.

18.2.17.15 Modification of Automatic Depressurization System Logic--Feasibility for Increased Diversity for Some Event Sequences (II.K.3.18)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.16 Interlock on Recirculation Pump Loops (II.K.3.19)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.17 Restart of Core Spray and Low-Pressure Coolant-Injection Systems (II.K.3.21)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.18 Automatic Switchover of Reactor Core Isolation Cooling System Suction--Verify Procedures and Modify Design (II.K.3.22)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.19 Confirm Adequacy of Space Cooling for High- Pressure Coolant Injection and Reactor Core Isolation Cooling Systems (II.K.3.24)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.20 Effect of Loss of Alternating-Current Power on Pump Seals (II.K.3.25)

18.2.17.20.1 NRC Guidance Per NUREG-0737

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Clarification

The intent of this position is to prevent excessive loss of reactor coolant system (RCS) inventory following an anticipated operational occurrence. Loss of ac power for this case is construed to be loss of offsite power. If seal failure is the consequence of loss of cooling water to the reactor coolant pump (RCP) seal coolers for 2 hours, due to loss of offsite power, one acceptable solution would be to supply emergency power to the

component cooling water pump. This topic is addressed for Babcock and Wilcox (B&W) reactors in Section II.K.2.16.

18.2.17.20.2 Union Electric Response

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals, and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from

the reactor coolant to the RCP internals. In the event of a loss of offsite power, the RCP motor is deenergized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and both seal injection flow and component cooling water to the thermal barrier heat exchanger are automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to a loss of seal cooling during a loss of offsite power for at least 2 hours.

18.2.17.20.3 Conclusion

The Callaway design meets the RCP seal cooling requirements of this section.

18.2.17.21 Provide Common Reference Level for Vessel Level Instrumentation (II.K.3.27)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.22 Verify Qualification of Accumulators on Automatic Depressurization System Valves (II.K.3.28)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.23 Study to Demonstrate Performance of Isolation Condensers with Noncondensibles (II.K.3.29)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.24 Revised Small-Break Loss-of-Coolant Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K (II.K.3.30)

18.2.17.24.1 NRC Guidance Per NUREG-0737

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC

approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Clarification

As a result of the accident at TMI-2, the Bulletins and Orders Task Force was formed within the Office of Nuclear Reactor Regulation. This task force was charged, in part, to review the analytical predictions of feedwater transients and small-break LOCAs for the purpose of assuring the continued safe operation of all operating reactors, including a determination of acceptability of emergency guidelines for operators.

As a result of the task force reviews, a number of concerns were identified regarding the adequacy of certain features of small-break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each lightwater reactor (LWR) vendor's models, were documented in the task force reports for each LWR vendor. In addition to the modeling concerns identified, the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small-break LOCA model as required by II.4 of Appendix K to 10 CFR 50 was needed. This included providing predictions of Semiscale Test S-07-10B and LOFT Test (L3-1) and providing experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small-break LOCAs.

Based on the cumulative staff requirements for additional small-break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small-break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small-break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision.

The specific staff concerns regarding small-break LOCA models are provided in the analysis sections of the B&O Task Force reports for each LWR vendor, (NUREG-0635, -0565, -0626, -0611, and -0623). These concerns should be reviewed in total by each holder of an approved emergency core cooling system (ECCS) model and addressed in the evaluation as appropriate.

The recent tests include the entire Semiscale small-break test series and LOFT Tests (L3-1) and (L3-2). The staff believes that the present small-break LOCA models can be both qualitatively and quantitatively assessed against these tests. Other separate effects tests (e.g., ORNL core uncovery tests) and future tests, as appropriate, should also be factored into this assessment.

Based on the preceding information, a detailed outline of the proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of the models, if any, the licensee intends to upgrade, (2) which areas the licensee intends to address by further justification of acceptability, (3) test data to be used as part of the overall verification/upgrade effort, and (4) the estimated schedule for performing the necessary work and submitting this information for staff review and approval.

18.2.17.24.2 Union Electric Response

The present Westinghouse Small Break Evaluation Model used to analyze the Callaway Plant (refer to Section 15.6.5) is in conformance with 10 CFR Part 50, Appendix K. However (as documented in Ref. 6), Westinghouse has addressed the specific NRC items contained in NUREG-0611 in a model (NOTRUMP) documented in WCAP 10054 (dated 12/28/82) and WCAP 10079 (dated 11/12/82). The NRC approved NOTRUMP as satisfying II.K.3.30 in a safety evaluation dated May 21, 1985. NOTRUMP was also found to be in full compliance with 10 CFR 50, Appendix K and was designated as the new Westinghouse licensing tool for small-break LOCA evaluations to satisfy the provisions of II.K.3.31.

18.2.17.24.3 Conclusion

The NOTRUMP code satisfies the provisions of NUREG-0737 Item II.K.3.30.

18.2.17.25 Plant-Specific Calculations to Show Compliance With 10 CFR Part 50.46 (II.K.3.31)

18.2.17.25.1 NRC Guidance Per NUREG-0737

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs), as described in item II.K.3.30 to show compliance with 10 CFR 50.46, should be submitted for NRC approval by all licensees.

18.2.17.25.2 Union Electric Response

The present Westinghouse Small Break Evaluation Model and small break LOCA analyses for Callaway Plant (refer to Section 15.6.5) are in conformance with 10 CFR Part 50, Appendix K and 10 CFR Part 50.46. As stated in the response to Item II.K.3.30 (refer to Section 18.2.17.24.2), Westinghouse has addressed the specific NRC items contained in NUREG-0611 in a model change documented in WCAP 10054 (dated 12/28/82) and WCAP 10079 (dated 11/12/82). On May 21, 1985, the NRC approved the new Westinghouse small break LOCA model, NOTRUMP, for use in satisfying the TMI Action Item II.K.3.30. On November 15, 1985, Union Electric informed the NRC (Reference 14) that a revised Callaway LOCA analysis was being submitted as

part of the Cycle Two Licensing Submittal. The revised analysis would utilize the NOTRUMP code. The use of this code satisfied the requirements of TMI Action Item II.K.3.31.

By letter dated October 16, 1986 (Reference 17), the NRC concluded that the generic study results could be used to resolve NUREG-0737, Item II.K.3.31.

18.2.17.25.3 Conclusion

Based on the above discussion, the requirements of NUREG-0737, Item II.K.3.31 are met.

18.2.17.26 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure (II.K.3.44)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.27 Evaluation of Depressurization with Other than Automatic Depressurization System (II.K.3.45)

Not applicable to Westinghouse pressurized water reactors.

18.2.17.28 Identify Water Sources Prior to Actuation of Automatic Depressurization System (II.K.3.57)

Not applicable to Westinghouse pressurized water reactors.

18.2.18 REFERENCES

1. Letter OG-57, dated April 20, 1981, Jurgensen, R. W. (Chairman, Westinghouse Owners Group) to Check, P. S. (NRC).

2. Wood, D. C. and Gottshall, C. L., "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants," WCAP-9804, February 1981.

3. "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," WCAP-9584 (Proprietary) and WCAP-9585 (Non-Proprietary), August 1979.

4. Letter OG-49, dated March 3, 1981, Jurgensen, R. W. (Chairman, Westinghouse Owners Group) to Ross, D. F., Jr. (NRC).

5. Letter OG-50, dated March 23, 1981, Jurgensen, R. W. (Chairman, Westinghouse Owners Group) to Ross, D. F., Jr. (NRC).

6. Letter NS-TMA-2318, dated September 26, 1980, Anderson, T. M. (Westinghouse) to Eisenhut, D. G. (NRC).

7. Letter NS-TMA-2357, dated December 23, 1980, T. M. Anderson (Westinghouse) to D. G. Eisenhut (NRC).

8. Rockwell, T., Reactor Shielding Design Manual, D. Van Nostrand Co., New York, New York, 1956.

9. QAD-CG: A Combinatorial Geometry Version of QAD-P5A, Bechtel Power Corporation internal computer code.

10. Letter SLNRC 83-0048, dated September 1, 1983, N. A. Petrick (SNUPPS) to H. R. Denton (NRC).

11. Letter SLNRC 84-004, dated January 16, 1984, N. A. Petrick (SNUPPS) to H. R. Denton (NRC).

12. Letter SLNRC 83-002 (distributed as 82-002), dated January 7, 1983, N. A. Petrick (SNUPPS) to H. R. Denton (NRC).

13. "A Generic Assessment of Significant Flaw Extension, Including, Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Power Plants," WCAP 10319, December 1983.

14. Letter ULNRC-1207, dated November 15, 1985, R. J. Schukai (UE) to H. R. Denton (NRC).

15. Letter NS-NRC-86-3099, dated February 20, 1986, Rahe, E.P. (Westinghouse) to Taylor, J. M. (NRC).

16. "Methodology for Qualifying Westinghouse WRD-Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Rev. 6-A, dated November 1983.

17. Safety Evaluation Report, WCAP-11145, Westinghouse Small-Break LOCA ECCS Evaluation Model Generic Study with NOTRUMP Code, transmitted by NRC letter (C. Rossi) to Westinghouse Owners Group (L. Butterfield), dated October 6, 1986.

TABLE 18.2-2 ESSENTIAL/NONESSENTIAL CONTAINMENT PENETRATIONS

Fig. 6.2.4-1, Sheet	Penetration	Service	Essential/ Nonessential
1	P-1	Main steam/PORV	Nonessential /essential
2	P-2	Main steam/PORV	Nonessential /essential
3	P-3	Main steam/PORV & AFW steam	Nonessential /essential
4	P-4	Main steam/PORV & AFW steam	Nonessential /essential
5	P-5	Main/aux. feedwater	Nonessential /essential
6	P-6	Main/aux. feedwater	Nonessential /essential
7	P-7	Main/aux. feedwater	Nonessential /essential
8	P-8	Main/aux. feedwater	Nonessential /essential
9	P-9	SG blowdown	Nonessential
10	P-10	SG blowdown	Nonessential
11	P-11	SG blowdown	Nonessential
12	P-12	SG blowdown	Nonessential
13	P-13	Containment recirculation sump suction to containment spray pump	Essential
14	P-14	Containment recirculation sump suction to RHR pump	Essential
15	P-15	Containment recirculation sump suction to RHR pump	Essential
16	P-16	Containment recirculation sump suction to containment spray pump	Essential
17	P-21	RHR hot leg injection	Essential

TABLE 18.2-2 (Sheet 2)

Fig. 6.2.4-1, Sheet	Penetration	Service	Essential/ Nonessential
18	P-22	RCP-B seal water supply	Essential
19	P-23	CVCS letdown	Nonessential
20	P-24	RCP seal water return	Nonessential
21	P-25	Reactor makeup water supply	Nonessential
22	P-26	Reactor coolant drain tank discharge	Nonessential
23	P-27	RHR cold leg injection loops 3 and 4	Essential
24	P-28	ESW supply to containment air coolers	Essential
25	P-29	ESW return from containment air coolers	Essential
26	P-30	Instrument air supply	Nonessential
27	P-32	Containment sump pump discharge	Nonessential
28	P-34	Containment ILRT test line	Nonessential
29	P-36	Maintenance Spare	Nonessential
30	P-39	RCP-C seal water supply	Essential
31	P-40	RCP-D seal water supply	Essential
32	P-41	RCP-A seal water supply	Essential
33	P-43	Auxiliary steam supply - decontamination	Nonessential
34	P-44	Reactor coolant drain tank vent	Nonessential
35	P-45	Accumulator nitrogen supply	Nonessential
36	P-48	SI pump-B, discharge to hot legs 1 and 4	Essential
37	P-49	SI pumps to cold legs 1, 2, 3, and 4	Essential
38	P-50	Maintenance Spare	Nonessential

TABLE 18.2-2 (Sheet 3)

Fig. 6.2.4-1, Sheet	Penetration	Service	Essential/ Nonessential
39	P-51	ILRT pressure sensing lines	Nonessential
40	P-52	RHR shutdown suction	Essential
41	P-53	Fuel pool cooling and cleanup, refueling pool supply	Nonessential
42	P-54	Fuel pool cooling and cleanup, refueling pool suction	Nonessential
43	P-55	Fuel pool cooling and cleanup, refueling pool skimmer suction	Nonessential
44	P-56	Post-LOCA hydrogen analyzer return	Essential
45	P-57	Nuclear Sampling System	Nonessential
46	P-58	Accumulator fill line from SI pump	Nonessential
47	P-59	RVLIS Sample Line	Nonessential
48	P-62	Pressurizer relief tank nitrogen supply	Nonessential
49	P-63	Service air supply	Nonessential
50	P-64	Nuclear Sampling	Nonessential
51	P-65	Hydrogen purge	Nonessential
52	P-66	Containment spray supply pump B	Essential
53	P-67	Fire protection supply	Nonessential
54	P-68	Maintenance Spare	Nonessential
55	P-69	Pressurizer vapor sample	Nonessential
56	P-71	ESW supply to containment air coolers	Essential
57	P-73	ESW return from containment air coolers	Essential
58	P-74	CCW supply	Essential

TABLE 18.2-2 (Sheet 4)

Fig. 6.2.4-1, Sheet	Penetration	Service	Essential/ Nonessential
59	P-75	CCW return	Essential
60	P-76	CCW return RCP thermal barrier	Essential
61	P-78	S.G. drain	Nonessential
62	P-79	RHR shutdown suction	Essential
63	P-80	CVCS charging	Nonessential
64	P-82	RHR discharge to hot legs loops 1 and 2	Essential
65	P-83	S.G. D sample	Nonessential
66	P-84	S.G. A sample	Nonessential
67	P-85	S.G. B sample	Nonessential
68	P-86	S.G. C sample	Nonessential
69	P-87	SI pump A discharge to hot legs loops 2 and 3	Essential
70	P-88	Boron injection supply to cold legs loops 1, 2, 3, and 4	Essential
71	P-89	Containment spray supply pump A	Essential
72	P-91	RVLIS Sample	Nonessential
73	P-92	ECCS test line return	Nonessential
74	P-93	R.C. loop and pressurizer liquid samples	Nonessential
75	P-95	Accumulator tank sample	Nonessential
76	P-97	Post-LOCA hydrogen analyzer return	Essential
77	P-98	Breathing Air	Nonessential
78	P-99	Post-LOCA hydrogen analyzer supply	Essential

TABLE 18.2-2 (Sheet 5)

Fig. 6.2.4-1, Sheet	Penetration	Service	Essential/ Nonessential
79	P-101	Post-LOCA hydrogen analyzer supply	Essential
80/81	P-103/104	Containment pressure sensing monitors	Essential
82	P-160	Containment purge exhaust	Nonessential
83	P-161	Containment purge supply	Nonessential
84	E-256	Containment pressure transmitters	Essential

TABLE 18.2-3 DETAILS FOR THE THERMOCOUPLE/CORE COOLING MONITOR SYSTEM

<u>Display</u>

Information Displayed (T-Tsat, Tsat, Press, etc.)	P-Psat subcooled- T-Tsat - superheated
Display Type (analog, digital, display)	Analog (control board), digital (electronics package), Display (Plant Computer)
Continuous or on Demand	Continuous (control board) and on demand (electronics package)
Single or Redundant Display	Redundant
Location of Display	Control board and control room
Alarms (include set points)	Caution: 25°F subcooled for RTD 15°F subcooled for T/C
	Alarm: 0°F subcooled for RTD and T/C
Overall uncertainty (F, psi)	Digital: 4°F for T/C, 3°F for RTD
	Analog: 5°F for T/C, 5°F for RTD
Range of Display	Calibrated: 200°F subcooled to 2000°F superheat
	Overall: Never off scale
Qualifications (seismic, environmental, IEEE 323)	Seismic and environmental

<u>Calculator</u>

Type (process computer, dedicated digital or analog calc.)	Dedicated digital
If process computer is used specify availability (percent of time)	NA
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	Auctioneered high hot leg RTD or average incore thermocouple. Auctioneered low reactor coolant pressure

TABLE 18.2-3 (Sheet 2)

Qualifications (seismic, environmental, IEEE 323)	Seismic and environmental
Calculational Technical (steam tables, functional fit, ranges)	Functional fit - ambient to critical point
<u>Input</u>	
Temperature (RTDs or T/Cs)	RTDs, T/Cs, and Tref
Temperature (number of sensors and locations	RTDs - 2 hot leg and 2 cold leg/channel T/Cs - 25 per channel
Range of temperature sensors	RTDs - 0-700°F T/Cs - 0-2500°F Calibration unit range - 0-2500°F
Uncertainty* of temperature sensors (F at 1)	See WCAP 8587
Qualifications (seismic, environmental, IEEE 323)	Seismic and environmental
Pressure (specify instrument used)	Qualified Pressure Transmitter
Pressure (number of sensors and locations)	1 wide range - RCS loop 2 narrow range - pressurizer
Range of pressure sensors	Wide range 0-3000 psig narrow range - 1700-2500 psig
Uncertainty* of pressure sensors (psi at 1)	See WCAP 8587
Qualifications (seismic, environmental, IEEE 323)	Seismic and environmental

* Uncertainties must address conditions of forced flow and natural circulation.

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 to the Callaway Plant FSAR Site Addendum but is now maintained as a separate licensing document as referenced in Appendix 13.3A of the FSAR. A successful emergency response exercise to test the integrated capability and major portions of the basic elements existing within the emergency preparedness plans and organizations was performed in March 1984. Radiological Emergency Response Exercises are held in accordance with the RERP.

18.3.1.3 Conclusion

Union Electric has provided the NRC with documentation relative to the emergency planning activities at Callaway which satisfies the requirements of 10CFR50, Section 50.47 and Appendix E, and the supplementary NRC guidance in NUREG-0654.

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. <u>Location</u>

    It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information 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 capability 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, during an emergency. As a minimum, data (historical in addition to current status) should 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 for:

1) Reactor Coolant System
2) Containment
3) Effluent Treatment
4) Release Paths

c. Offsite Radiological

1) Meteorology
2) Offsite Radiation Levels

8. <u>Data Transmission</u>

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. <u>Structural Integrity</u>

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 establish 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 follow the guidance of NUREG-0696, Final Report, entitled "Functional Criteria for Emergency Response Facilities", to the extent described in the following portions of Section 18.3.2.2. Standard engineering practices were used for the design, manufacturing, and construction of these facilities and equipment, as described in SLNRC 81-38, dated June 1, 1981.

NRC Generic Letter 82-33, dated December 17, 1982 provided guidance for meeting regulatory requirements for, among other issues, NUREG-0737, Item III.A.1.2. Based on the NRC review of the response to Generic Letter 82-33 (SLNRC 83-19, dated April 15, 1983), a license condition (License Condition C(7)(b)) was issued which required that the TSC and EOF be operational prior to startup following the first refueling outage at Callaway Plant. This license condition was closed out by ULNRC-1288, dated April 8, 1986.

Technical Support Center

The Technical Support Center (TSC) at the Callaway Site is located within the protected area and near the on-site buildings that contain the offices of managerial, engineering and the plant support personnel. The location of the Technical Support Center is shown in the RERP.

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 personnel designated to man the TSC have their offices nearby.
- There is ready accessibility to plant data available in the Service Building which is not stored in the TSC (e.g., vendor manuals).

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

The TSC is a one story building located at grade level. The walls are reinforced concrete 8 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 there is working space that contains displays of plant status, meeting and discussion areas, communications equipment, and document storage. Additional areas within the TSC are occupied by a mechanical equipment room, which contains HVAC equipment, a standby diesel-generator for the TSC, and limited toilet and kitchen. This is sufficient space for at least 25 persons, including five NRC personnel. For any extended duration of TSC operation, additional toilet, locker room, and kitchen facilities in the Service Building are available.

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

Radiation monitoring in the TSC is performed using electronic dosimeters. These designated electronic dosimeters have a range of 0 to 100,000 mR/hr, and provide an alarm function based on a pre-designated set point.

Airborne radioactivity in the TSC is monitored using a continuous air monitor. This monitor is located in an area common to TSC inhabitants. The monitor provides an indication of airborne radioactive material collected on a particulate filter. The continuous air monitor has an alarm function. A portable air sampler would be used to obtain a grab sample when the continuous air sampler is in alarm. Air sampler filter media would be analyzed to determine the level of particulate and radioiodine concentration.

Electric power to the TSC in a post-accident situation is normally provided by a transformer from off-site power. Alternatively, there is a standby diesel generator, rated at 288 kVA, that is started manually utilizing dedicated battery power. The diesel generator has sufficient capacity to power all TSC loads, including plant computer terminals, communications equipment, HVAC and lighting. In addition, the diesel generator has sufficient capacity to power selected loads in the Service Building Computer Room, in the event power is lost to the Service Building during normal operation or under post-accident conditions.

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. Selected plant computer terminals and communications systems have uninterruptable power supplies. Emergency lighting consisting of self-contained battery units is also provided in the TSC.

Protective clothing, respirators, and personnel radiation monitors to permit up to 10 persons to function within radiation areas are accessible to the TSC.

The conditions for manning the TSC are described in general terms in the Callaway Radiological Emergency Response Plan. Detailed procedures have been developed, as Emergency Plan implementing procedures.

Operations Support Area

The Operations Support Area (OSA) is described in the Radiological Emergency Response Plan. The OSA provides ample space for assembly of personnel and has communications with various other emergency facilities. Equipment, tools and protective clothing are also available.

Emergency Operations Facility

At Callaway the Emergency Operations Facility (EOF) is located approximately 1 mile from the plant, as discussed in the RERP. It is a one-story building of 13,000 square feet.

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, Selected Plant computer terminals and communications equipment are provided with an Uninterruptible Power Supply. Emergency lighting consisting of self-contained battery units is also provided in the EOF.

In the event the EOF becomes uninhabitable, a backup EOF will be established in the State of Missouri Emergency Operations Center, Jefferson City, MO.

Jefferson City is located approximately 25 miles southwest of the plant site.

Technical Data

The Emergency Response Facility Information System (ERFIS) is an application of the Plant Computer. System displays, individual data points displays, trends and historical values are available for all major plant systems. ERFIS displays and printers are available in the TSC and EOF.

The Plant computer provides radiological and meteorological data as well as trending capabilities. Displays are located in the Control Room, TSC, Health Physics Access Area, EOF and Plant Computer Room.

The Safety Parameter Display System is available to aid operators in the rapid detection of abnormal operating events. SPDS displays are available in the Control Room, TSC, and EOF.

Task Functions for the TSC and EOF

Refer to Section 18.3.1.2

18.3.2.3 Conclusion

The functional description of each emergency response facility described above details how Union Electric meets the appropriate NRC guidance.

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

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 radiological 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 Preparedness in Support of Nuclear Power Plants."

Clarification

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 Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides detailed items to be included in the upgraded emergency plans and, along with the revised rules, 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 plans 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 position 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 Support 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 for use in the assessment of the offsite consequences of a radiological 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 system should allow the eventual incorporation of effuent monitoring 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.

18.3.4 INTEGRITY OF SYSTEMS OUTSIDE OF CONTAINMENT (III.D.1.1)

18.3.4.1 NRC Guidance Per NUREG-0737

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

  (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

  (b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical

levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside the containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

(1) Systems that should be leak tested are as follows (any other plant system which has similar functions or postaccident characteristics, even though not specified herein, should be included):

Residual heat removal (RHR)

Containment spray recirculation

High-pressure injection recirculation

Containment and primary coolant sampling

Reactor core isolation cooling

Makeup and letdown (PWRs only)

Waste gas (includes headers and cover gas system outside of the containment in addition to decay or storage system)

Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

(2) Testing of gaseous systems should include helium leak detection or equivalent testing methods.

(3) Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

18.3.4.2 Union Electric Response

This defines Union Electric's program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems considered include the

recirculation portion of the containment spray system, safety injection system, chemical and volume control system, and RHR system. The program is as follows:

1. Preventive maintenance and periodic visual inspection requirements:

   PROGRAM: One aspect of the P.M. program is the periodic replacement of valve packings, pump seals, and flange gaskets based on service life. This service life is based on both maintenance history and visual inspections.

   Periodic walkdowns of accessible systems are performed. One objective of these walkdowns is to identify and correct leakage both from recirculation systems and other secondary systems.

   Operations, via their normal duties and responsibilities, also walk-down and visually inspect accessible systems. Identification and correction of leakage is one of the criteria specified for these walkdowns.

2. Integrated leak test requirements for each system at refueling cycle intervals or less:

   PROGRAM: Callaway is committed to ASME Section XI for commercial operation. Section XI requires VT-2 examinations on Class 2 systems during each inspection period. For Callaway, this amounts to a VT-2 visual examination for leakage on all recirculation systems every 3 to 4 years.

   In addition, Operations performs leakage testing on all recirculation systems with systems at operating pressure. This is accomplished per written procedure on a refueling frequency.

A description of the Union Electric program was provided by ULNRC-693, dated December 2, 1983.

18.3.4.3 Conclusion

The Callaway design includes provisions to insure the integrity of fluids systems which are postulated to contain highly contaminated fluids following a design basis accident. The provision is based on the preservice and inservice tests required by the ASME Code. These provisions provide assurance that these systems will perform their

intended functions, including leaktightness, following a design basis accident. This commitment satisfies Item III.D.1.1 of NUREG-0737.

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.

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.

18.3.5.2 Union Electric Response

Equipment and associated training and procedures are available to enable the accurate determination of airborne radioiodine concentrations in areas within the facility where plant personnel may be present during an accident. Silver Zeolite cartridges will be used for radioiodine sampling. Typically samples will be analyzed using a gamma spectroscopy system. Capabilities exist to remove interfering gaseous activity from the cartridge by purging with clean, noble gas free air or nitrogen or by placing the cartridge in a vacuum dessicator. Gross activity determination may be performed using a count rate meter, without purging, for a rapid indication of radioiodine concentrations. If high background precludes the use of the counting facilities on site, backup facilities are available at the University of Missouri, Research Reactor.

18.3.6 CONTROL ROOM HABITABILITY (III.D.3.4)

18.3.6.1 NRC Guidance per NUREG-0737

Position

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall ensure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Clarification

(1) All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

(2) All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity,
2.2.3	Evaluation of Potential Accidents, and
6.4	Habitability Systems

shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

a. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release";

b. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"; and,

c. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Licensees shall submit the results of their findings as well as the basis for those findings by Janaury 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1, to NUREG 0737, item III.D.3.4 is provided.

(3) All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references.

These licensees shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident (LOCA) containment leakage and engineered safety feature (ESF) leakage contribution outside the containment, as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i.e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will

operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

18.3.6.2 Union Electric Response

The safety design bases for the habitability system for the control room are defined in Section 6.4. This section also discusses the applicable recommendations of Regulatory Guides 1.78, and 1.95. The results of dose calculations for a design basis loss-of-coolant accident release are presented in Section 15.6.5 and 15A.3.

The design of the habitability system for the control room envelope meets the appropriate recommendations of Regulatory Guide 1.78 and 1.95 and requirements of GDC 19.

18.3.6.3 Conclusion

The design of the control room habitability system meets the recommendations of item III.D.3.4 of NUREG-0737.