TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

September 14, 1981

RECEIVED SEP 16 1981 U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Enclosed are 10 copies of TVA responses for Watts Bar Nuclear Plant to NRC concerns as specified in NUREG-0737.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 14th day of Sept. 1981

Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

Enclosure

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ENCLOSURE

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

TVA RESPONSES TO NUREG-0737

September 11, 1981

I.A.1.1 SHIFT TECHNICAL ADVISOR

NRC Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor functon 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 STA's that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Changes to Previous Requirements and Guidance

There have been no changes to the requirements resulting from NUREG-0660 and the October 30, 1979, letter from H. R. Denton to All Operating Nuclear Power Plants.

NRC Clarification

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

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979, letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980, letter (as revised by Item I.A.1.3 of Enclosure 3 to NUREG-0737), 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 STA's. These efforts include work on the revision to ANS-3.1, work by the Institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO has 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 included as Appendix C to NUREG-0737. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STA's. The NRC staff has found that the descriptions as 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. 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.)

All applicants for operating licensees shall provide a description of their STA training program and their plans for requalification training. This description shall demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter.

All applicants for operating licensees shall provide 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.)

<u>Implementation</u>

(1) Training that meets the lessons learned requirements shall be completed by the time the fuel-loading license is issued.

(2) A description of the current training program and demonstration of conformance with the October 30, 1979 letter shall be submitted on a schedule consistent with review schedule for applicants for operating licenses.

(3) A description of the long-term STA program shall be submitted on a schedule consistent with review schedule for applicants for operating licenses.

<u>Type of Review</u>

Applicants for operating licenses will be reviewed as part of the licensing review.

<u>Documentation Required</u>

Documentation will be required as noted above.

<u>Technical Specification Changes Required</u>

Changes to technical specifications will be required.

<u>References</u>

NUREG-0578, Recommendation 2.2.1.b

NUREG-0660

INPO Document, see Appendix C to NUREG-0737

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

Letter from D. G. Eisenhut, NRC, to All Licensees and Applicants, dated July 31, 1980.

TVA RESPONSE

The shift technical advisor requirements will be implemented upon receipt to an operating license.

TVA is providing an on-shift technical advisor to the shift supervisor to support the diagnosis of off-normal events and to advise the shift supervisor of actions to terminate or mitigate the consequences of such events.

The Shift Technical Advisor will have the following qualifications: (1) training in basic engineering principles, (2) extensive training in plant transient and accident response, (3) technical specification training with emphasis on the basis for limiting conditions for operation, and (4) significant reactor training on systems and operating procedures.

The duties of the Shift Technical Advisor will include: (1) control room support in the diagnosis of off-normal events, (2) advice to the shift supervisor to terminate or mitigate the consequences of off-normal events, (3) engineering evaluations of plant conditions required for maintenance and testing, and (4) cognizant of current information disseminated by TVA's operating experience review group.

The Shift Technical Advisor training program will cover the following subjects as a minimum:

1. Nuclear Plant Systems

    A. Basic Components
    B. Reactor Coolant System
    C. Emergency Core Cooling Systems
    D. Residual Heat Removal Systems
    E. Containment Systems
    F. Control Rod Drive Systems
    G. Fuel Handling Systems
    H. Secondary Side and Auxiliary Systems

2. Power Plant Operation

    A. Startup
    B. Shutdown
    C. Power Operation
    D. Integrated System Response

3. Transients and Accidents

    A. Licensing Basis Transients and Accidents

        1. Assumptions
        2. Conservatisms
        3. Minimum equipment taken credit for

B. Transient and Accident Recognition and Operator Action

1. FSAR Chapter 15 events
2. Instrumentation failures
3. Degraded conditions of system availability

4. Limiting Conditions for Operation

A. Technical Specification Definition
B. Technical Specification Bases

5. TVA Operational Practices

A. Job Assignments and Responsibilities
B. TVA Emergency Plan
C. Document Familiarization
D. Clearance Procedures
E. Plant Safety Practices and Procedures

TVA believes that the STA must have a basic knowledge of fundamental plant operation to be able to fulfill his advisor function during abnormal events.

In addition to the accident assessment function, the shift technical advisor will be cognizant of information determined by the TVA Operating Experience Review Group. The shift technical advisor will be independent of duties that detract from his primary functions or dilute his dedication to these primary functions. The shift technical advisor will be an addition to the previously defined operating staff.

Organizationally, the STA will be a full-time shift employee who will work for the plant Reactor Engineer; thus, maintaining independence from the operations staff and will be available within 10 minutes of being summoned during the shift. The STA will have an advisory role only. The ultimate responsibility for plant operations during normal and abnormal events must rest with the shift supervisor.

More specific information concerning the qualification, responsibilities, and duties of the STA are provided in TVA Division Procedures Manual (DPM No. N79A15). This information was provided to the NRC during the license review of Sequoyah Nuclear Plant by letter dated February 17, 1980, from L. M. Mills to L. S. Rubenstein.

The long-term program for phase out of the STA program by TVA has not been finalized.

I.A.1.2 SHIFT SUPERVISOR RESPONSIBILITIES

NRC Position

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

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

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

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

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

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

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

NRC Clarification

The attachment provides clarification to the above position.

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.a)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V. P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANSI3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V.P. for Operations.

Implementation

Applicants must provide this information four months prior to the scheduled Safety Evaluation Report (SER).

Documentation Required

Documentation will be required as noted above.

Technical Specification Changes Required

None.

References

NUREG-0578, Recommendation 2.2.1.a.

TVA Response

The requirements are to be implemented by fuel loading for Watts Bar units 1 and 2.

1. The duties of the shift supervisor, as discussed in NUREG-0578, are performed by the assistant shift engineer. The V. P. for Operations is the Manager of Power Operations. TVA's administrative procedures, shift supervisor job descriptions, and training programs emphasize the primary management responsibility of the shift engineer. In addition, periodic retraining acts to reinforce his command responsibilities. While these existing measures provide a high level of confidence that the shift supervisor has primary management responsibility for safe operation of the plant, TVA will periodically issue a management directive which emphasizes this assignment of responsibility.

2a. Plant administrative procedures have been reviewed to ensure that they clearly define the authority and responsibilities of each position on shift. The duties and responsibilities of the shift supervisor, as specified in the job description, are consistent with position statement 2a. Administrative instruction, the shift supervisor's (assistant shift engineer's) responsibilities, and the Watts Bar standard practices show TVA's current training program.

2b. The shift crew in TVA plants consists of the following: (1) a shift engineer who has an SRO license and who has overall responsibility for the plant when higher level 'in-line' management employees are not present, (2) an assistant shift engineer (also has an SRO license) for each unit who has supervisory responsibility for all normal, abnormal, and emergency activities on his assigned unit, (3) a unit operator (with an RO license) for each unit, and (4) other employees as appropriate. The duties of the shift supervisor as discussed in NUREG-0578 and -0737 are performed by the assistant shift engineer on each unit. For purposes of our responses, we will use the term assistant shift engineer for shift supervisor.

The assistant shift enigneer's normal work station is in the control room, but he periodically makes inspections of plant equipment. We will immediately go to the control room during emergency situations.

He remains in the control room at all times during accident situations to direct the activities of the unit operator unless formally relieved of this function by the shift engineer. The shift engineer may, in turn, be formally relieved by the assistant operations supervisor or the operations supervisor (both also hold an SRO license).

2c. In the event that the assistant shift engineer (shift supervisor) is absent, the unit operator will be the lead operator on the unit to which he is assigned. For multiple-unit plants, an additional licensed operator will be available in the control complex to act as an assistant to the unit operator in abnormal or emergency situations. The line of command is clearly specified in administrative procedures.

3. The shift engineer and assistant shift engineers will receive such training.

4. The administrative duties of the shift supervisor have been reviewed by the senior officer of TVA responsible for plant operations. Administrative functions that detract from or are subordinate to ensuring safe operation of the plant will be assigned to other employees. The following actions have already been taken:

   1. A clerk has been assigned to the shift engineer's office on each shift to perform administrative details formerly done by the shift engineer.

   2. Part of the routine 'non-management' duties of the assistant shift engineer have been assigned to other employees.

I.A.1.3 SHIFT MANNING

NRC 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.

Changes to Previous Requirements and Guidance

Errors were discovered in the last column of the table attached to the letter of July 31, 1980. A corrected table is enclosed page I.A.1.3-4 of NUREG-0737; a bar in the margin indicates the correction.

The overtime requirements have been rewritten to be more flexible.

NRC Clarification

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

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

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians, and key maintenance personnel).

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

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance, or major plant modifications), the following overtime restrictions should be followed:

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

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

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

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

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

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

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

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

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

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.

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

Implementation

(1) Overtime administrative procedures shall be established by fuel loading for applicants for operating licenses.

(2) Staffing requirements shall be completed by fuel load for operating license applicants.

Type of Review

Applicants for operating licenses will be reviewed prior to implementation.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specifications will be required for minimum shift crew manning.

References

NUREG-0660

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

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

TVA RESPONSE

TVA will meet the requirements for shift manning and overtime for operation of Watts Bar Nuclear Plant Units 1 and 2. The Watts Bar technical specifications will list the minimum shift crew required for operation. Overtime administrative procedures will be established before fuel loading.

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

NRC Position

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

Changes to Previous Requirements

Changes to the previous requirements will permit various paths to provide experience equivalent to 1 year's experience as a licensed operator.

NRC 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 SRO's. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation.

In addition, in order to attract degree-holding engineers to consider the shift supervisor's job as part of their career development, NRC 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 assures 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 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.

Applicability

This requirement applies to all applicants for operating licenses (after initial criticality).

Implementation

This requirement applies to applicants for senior reactor operator licenses received after December 1, 1980.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required from the facility. Information will be contained in individual applications.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

TVA RESPONSE

As specified in FSAR Section 13.2 and Watts Bar Nuclear Plant standard practices, TVA's program for licensing of reactor operators and senior reactor operators for Watts Bar Nuclear Plant units 1 and 2 meets these requirements.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

NRC Position

Pending accreditation of training institutions, applicants for operating licenses will assure 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements included in the letter of March 28, 1980 from H. R. Denton to All Power Reactor Applicants and Licensees.

NRC 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 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 sit for an SRO examination.

Implementation

Applications for SRO examinations should be submitted. All applicants for operating license should submit documentation 2 months prior to the expected issuance of an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

TVA RESPONSE

I.A.2.3 Administration of Training Programs

TVA is complying with NUREG-0737 administration of training requirements for reactor operators and senior reactor operators for Watts Bar units 1 and 2. This program is identical to the Sequoyah program reviewed and approved by NRC. No changes to the programs are presently planned.

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS--SIMULATOR EXAMS (ITEM 3)

NRC Position

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

NRC Clarification

The clarification provides additional preparation time for utility companies and NRC to meet 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. 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.

Applicability

This requirement applies to all applicants for operator and senior operator licenses at power reactors.

Implementation

The schedule for applicants for operating license with simulators is prior to fuel load including cold examination.

Type of Review

No review will be performed. Arrangements will be made during the normal scheduling of examinations.

Documentation Required

No documentation is required. Arrangements will be made during the normal scheduling of examinations.

Technical Specification Changes Required

Changes to technical specifications will not be requried.

Reference

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS-SIMULATOR EXAMS

TVA RESPONSE

TVA has a simulator essentially the same as the Watts Bar Nuclear Plant units 1 and 2 control rooms for operator examinations. This satisfies the requirement described in NUREG-0737.

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP

NRC 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 anlaysis, 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements, however further guidance is provided in the 'Clarification' section that follows.

NRC 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 Guide 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 (STA's) 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, therfore, 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 utility's 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 see 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).

Applicability

This requirement applies to all applicants for operating license.

Implementation

This requirement shall be implemented prior to issuance of an operating license.

Type of Review

A preimplementation review will be performed.

Documentation Required

Each applicant for an operating license shall document in its application or amendments thereto, its plan for establishing and staffing the ISEG, including the qualifications of and the training to be given the ISEG staff.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

NUREG-0694, Item I.B.1.1 and Item I.B.1.2

TVA RESPONSE

TVA will establish an independent safety engineering group (ISEG). The ISEG will consist of a permanently assigned engineer and two part-time engineers onsite working in conjunction with a central office core staff. This central office staff will consist of five engineers working part-time to support the ISEG function. It is TVA's position that this satisfies the requirement of NUREG-0737.

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

NRC Position

In letters of September 27 and November 9, 1979 the Office of Nuclear Reactor Regulation required applicants for operating licenses to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also 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 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 (see 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, 39, 41, 47, 55, 57).

Changes to Previous Requirements and Guidance

A. Modification to Clarification

(1) Addresses owners' group and vendor submittals.

(2) References to task action plan Items I.C.8 and I.C.9.

(3) Scope of procedures review is explained.

(4) Establishes configuration control of guidelines for emergency procedures.

B. Modification to Implementation

(1) Deleted reference to NUREG-0578, Recommendation 2.1.9 for Item I.C.1(a)2, inadequate core cooling.

NRC Clarification

The letters of September 27 and November 9, 1979, required that procedures and operator training be developed for transients and accidents. The intiating 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 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 cooing would result.

Examples of multiple failure events include:

(1) Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator;

(2) Failure of main and auxiliary feedwater;

(3) Failure of high-pressure reactor coolant makeup system;

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

(5) Operator errors of omission or commission

The analyses should be carried out far enough into the event to assure 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:

(1) A detailed description of the methodology used to develop the guidelines;

(2) Associated control function diagrams, sequence-of-event diagrams,

or others, if used;

(3) The bases for multiple and consequential failure considerations;

(4) Supporting analysis, including a description of any computer codes used; and

(5) 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). This will involve review of the loss of coolant, steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequancy of Westinghouse guidelines will be identified to each NTOL during the emergency-procedure review.

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 NTOL'S under Item I.C.8 will be discontinued, and the review required for applications 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. 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 operaing 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 guidleine, 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 issue 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.

Implementation

Reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergnecy procedures should have been completed and submitted to the NRC for review by January 1, 1981. The NRC staff will review the analyses and guidelines and determine their acceptability by July 1, 1981, and will issue guidance to licensees on preparing emergency procedures from the guidelines. Following NRC approval of the guidelines, applicants for operating licenses issued prior to January 1, 1982, should revise and implement their emergency procedures at the first refueling outage after January 1, 1982. Applicants for operating licenses issued after January 1, 1982, should implement the procedures prior to operation. This schedule supersedes the implementation schedule included in NUREG-0578, Recommendation 2.1.9 for Item I.C.1(a)3, Reanalysis of Transients and Accidents.

Type of Review

A preimplementation review of guidelines will be performed.

A preimplementation review of procedures will be performed.

Documentation Required

See above, 'Implementation.'

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0578, Recommendation 2.1.9

NUREG-0660, Item I.C.8 and Appendix C, Tables C.1, C.2, C.3

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Plants, dated September 13, 1979

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated October 10, 1979

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979

TVA RESPONSE

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

TVA is a member of the Westinghouse Electric Corporation Owners' Group and will utilize the guidelines resulting from the owners' group work in preparation of the procedures for Watts Bar Nuclear Plant. The procedures and analysis work to date and a description of future efforts are specified in the attached copy of a letter from R. W. Jurgensen, of the Owner's Group, to S. H. Hanover (NRC) dated July 7, 1981

I.C.2 SHIFT AND RELIEF TURNOVER PROCEDURES

NRC Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

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

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

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

    (What to check and criteria for acceptable status shall be included on 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 ongoing auxiliary operators and technicians. Such checklists and logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or intiate 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 procedure (for example, periodic indepent verification of system alignments).

NRC Clarification

No clarification provided.

Implementation

The required procedure must be immplemented upon receipt of an operating license.

Documentation Required

None

Technical Specification Changes Required

None

References

NUREG-0578, Recommendation 2.2.1.C

SHIFT AND RELIEF TURNOVER PROCEDURES

TVA RESPONSE

TVA has developed and will implement shift and relief turnover procedures for Watts Bar units 1 and 2 which provide assurance that the oncoming shift possesses adequate knowledge of critical plant status information and system availability. A checklist or similar hard copy will be completed and signed by offgoing and oncoming shifts at each shift turnover.

This checklist includes critical plant parameters and allowable limits, availability and proper alignment of safety systems, and a listing of safety system components in a degraded mode along with the length of time in that mode. All shift personnel responsible for the status of critical equipment have relief checklists for oncoming and offgoing shifts that will include any core cooling equipment under maintenance or test that could degrade a safety system. In addition, a system will be established to evaluate the effectiveness of the turnover procedures.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITY

See Item I.A.1.2

I.C.4 CONTROL ROOM ACCESS

NRC Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to 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, and

2. Develop and implement procedures that establish a clear line of authoirty 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 of the control room.

NRC Clarification

No clarification provided.

Implementation

Upon receipt of an operating license.

Technical Specification Change Required

None

References

NUREG-0578 Recommendation 2.2.2.a

CONTROL ROOM ACCESS

TVA RESPONSE

TVA has developed and will implement plant specific administrative procedures that establish specific individual authority and responsibility as well as delineate various system or equipment functions related to controlling personnel access during normal and accident conditions. A control room access plan has been developed to provide direction to all members of the plant staff to ensure that those persons responsible for safe operation of the plant are able to perform effectively.

In addition, TVA has developed and will implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. These procedures clearly define the lines of communication and authority for plant management personnel and ensure that the shift supervisor, his assistant, or senior licensed management personnel are the only plant personnel who have the authority to direct licensed activities of licensed reactor operators.

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

NRC 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 assure 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, 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 assure 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) Assure 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 assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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 assure 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 and training and retraining programs.

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, and 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 assure 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 assure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealth with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. it is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

Implementation

Procedures governing feedback of operating experience to plant staff shall be completed and the procedures put into effect prior to issuance of an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item I.C.5

Letter from D. G. Eisenhut, NRC, to All Licensees, dated May 7, 1980.

PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

TVA RESPONSE

TVA will have a program and procedures in place at Watts Bar which will comply with NRC requirements outlined in NUREG-0737.

I.C.6 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

NRC Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necesary, to assure 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.

Changes to Previous Requirements and Guidance

Proposed requirement in NUREG-0660; this requirement is formally issued by this letter.

NRC 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 be reviewed and revised, as necessary, to assure 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 has prepared a draft revision to ANSI Standard N18.7-1972 (ANS 3.2) 'Adminsitrative 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 futuere, will endorse the latest draft revision to ANS 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 authoirty to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

(3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of

equipment control measures such as 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 systems 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 system 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.

Implementation

Applicants must review and revise procedures as necessary to reflect this position prior to fuel load.

Type of Review

A postimplementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0585, Recommendation 5

NUREG-0660, Item I.C.6, I.D.3

GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

TVA RESPONSE

Current plant administrative procedures require that:

(a) the alignment of all systems and components important to safety (see note 1) be verified prior to unit startup.

(b) changes in the alignment of any system important to safety be recorded on a system status sheet.

(c) shift personnel being relieved communicate information on any abnormal plant condition including temporary conditions.

(d) system operability be demonstrated before a system is returned to service, and

(e) approval by the shift supervisor or his representative be received prior to the performance of any activity on any systems important to safety or any activity that may affect systems important to safety. The shift supervisor or his representative is notified when any activity authorized to be performed on a system important to safety is completed or a change occurs in the scope of the activity.

Plant operating instructions require completion of a startup checklist prior to unit startup. This checklist is used to verify correct alignment of all systems important to safety. Anytime a critical component is changed from its normal position or condition, a system status sheet is completed and placed in a system status folder. Panel checklists are reviewed each shift to verify proper panel alignment exists for all systems important to safety. Panel checklists are reviewed each shift to verify that proper panel alignment exists for all systems important to safety.

It is TVA's opinion that this verification function can be performed adequately by an assistant unit operator (AUO) and that the use of licensed unit operators is not necessary. The AUO has sufficient training and familiarity with plant systems to ensure correct system alignment, and this policy will allow the licensed operator to remain the control room. The following list composes the systems for which second verification has been required.

A. Auxiliary Feedwater
B. Emergency Core Cooling
C. Emergency Gas Treatment
D. Essential Raw Cooling Water
E. Reactor Coolant System
F. Comonent Cooling Water
G. Containment Spray
H. Residual Heat Removal
I. Emergency Diesel Generators
J. Upper Head Injection

K. Spent Fuel Pit Cooling System

NOTES:

1 Equipment important to safety is defined as the reactor coolant system (pressure boundary components) and associated pressurizer and pressure relief systems, the residual heat removal system, engineered safety feature systems, engineered safety features electric power systems, and cooling water systems necessary to operate the above systems.

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

NRC Position

Applicant must have power ascension and emergency procedures reviewed by Westinghouse.

Implementation

This is an operating license requirement.

Technical Specification Change Required

None

References

NRC letter of July 26, 1980.

NSSS VENDOR REVIEW OF PROCEDURES

TVA RESPONSE

The Watts Bar Nuclear Plant power ascension and emergency procedures will be reviewed by Westinghouse and any required changes will be completed before fuel loading.

I.D.1 CONTROL ROOM DESIGN REVIEWS

NRC Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), 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 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 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC Clarification

NRR is presently developing human engineering guidelines to assist each licensee and applicant in performing a 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.' 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 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 the 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, 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 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.

Implementation

(1) Applicants for OL's:

Complete review, using NRC guidelines (NUREG-0700) issued in 1981, on a schedule that will be determined upon issuance of the guidelines.

(2) Applicants for OL's whose schedules do not permit a full review prior to licensing: Preliminary review complete and approved by NRC prior to issuance of the operating license.

Type of Review

A preimplementation review will be performed for operating license applicants.

Documentation Required

Applicants for OL's with impacted schedules should report on results of preliminary review prior to licensing.

Technical Specification Changes Required

Changes to technical specifications will not be required unless there are modifications to the control room.

References

NUREG-0660, Item I.D.1

NUREG/CR-1580 (Draft)

RESPONSE

The results of a preliminary design review of the Watts Bar Control Room was provided to the NRC by letter dated January 13, 1981, from L.M. Mills to A. Schwencer.

The following summarizes all the activities that have been accomplished to make up the preliminary review of Watts Bar (WBN).

The preliminary assessment of the WBN control room to identify significant human factors problems started on January 21, 1980. This initial two-day preliminary review revealed a number of areas that needed to be pursued. During the week of February 4, 1980, a preliminary control room review was accomplished on the Sequoyah Nuclear Plant (SQN) unit 1 control room prior to criticality. SQN unit 1 is similar (basically identical) to WBN units 1 and 2. A preliminary assessment conducted for either unit is appropriate for the other units.

This assessment was conducted by the Essex Corporation (under contract to the Nuclear Regulatory Commission, NRC) with a team of NRC and TVA personnel actively involved. Essex Corporation issued a report dated March 10, 1980 summarizing their findings. NRC identified from the report the significant items requiring immediate attention.

TVA provided corrections to these items, and they were documented in the Sequoyah Safety Evaluation Report dated September 4, 1980. These changes are also needed on WBN. They will be completed prior to each unit's fuel loading. The modifications are as follows:

1. Dedicated panel telephones will be installed to improve control room communications between operators.

2. Panel guardrails will be installed to prevent inadvertent actuation of switches close to the front edge of the main control panels. Also, red carpet will be installed at the base of vertical panels to designate off-limit areas to employees not performing a required task.

3. Arrangements will be made to maintain procedures in a specific location in the control room, and an index will be added to assist operators in locating specific emergency procedures. Also, immediate action steps in emergency procedures will be revised to eliminate references made to external documents.

4. Alarms important to safety will be arranged by priority by color coding annunciator windows.

5. Common panels containing controls and displays from multiple units will be modified by using color coding and adding specific unit numbering to provide unique identification of each control and display.

6. The bezels will be painted black on each overhead annunciator display panel to improve contrast between annunciator windows and background.

7. Carpet will be added to the control room to reduce background noise levels.

8. Control room procedures will be revised to instruct operators to use the lamp test buttons on the status monitoring panels to verify that a lamp is burned out, rather than implying that a system is unavailable.

TVA has continued the preliminary review of the WBN main control room by proceeding with internal design studies and task analyses of the control boards and by making trips to the plantsites and the plant simulators to identify human factor problems. These trips involved walk throughs of the operating instructions at both the Watts Bar plant and simulator. Detailed interviews were conducted with the operators at the plants and at the simulator with the instructors.

The control rooms were also examined to identify any significant human engineering deficiencies. The information obtained from these sources was reviewed with engineering design groups to determine the significant items and identify possible ways to implement the desired changes. These changes were then reviewed and coordinated with the plant personnel to finalize the changes to be incorporated.

The following is a list of items TVA has identified and the action to be taken:

1. Further steps will be taken to reduce the noise generated by the area radiation monitoring equipment located in the main control room.

2. Operating ranges will be added to the scale of indicating meters where possible in the main control room with the following criteria applied:

   a. <u>No color</u> (clear) - Normal operation.

   b. <u>Yellow</u> - Abnormal operating condition; with caution being taken and/or first alarm point of a two-level alarm.

   c. <u>Magenta</u> - Abnormal operating condition; action should be taken immediately and/or second alarm point of a two-level alarm.

3. The pointer on all indicating meters will be painted fluorescent orange.

4. All nametags will be reviewed to verify proper nomenclature is used. The addition of a nametag to handswitches to provide the power source supplying that device being controlled is under review.

5. The control boards will be modified using functional grouping and demarcation of related control elements using black graphic plastic for demarcation of systems. This will require the movement of approximately 34 devices.

6. The use of 475 functional nametags will be installed in coordination with the functional grouping.

7. Three compuer trend records will be moved from a back row panel (1M-10) to panels close to the operator on panels 1M-1.

8. The bezels of all meters will be changed to black to improve contrast of meter reading.

9. The test switches (25) located on panel 1M-6, which are infrequently used, are under study to be moved to back row panels. This will allow room for function grouping of devices and future additions on this panel, which is primarily made up of safety-related devices.

10. The proper scaling on indicators has been found to be a problem. Meters labeled in percent have been found to be a concern. There are 116 meters scaled in percent. A complete review of all meters is presently underway to determine proper units and ranges and to find meters that may require dual scaling (i.e., gallons and percent).

11. The nametags will be changed from black on grey to black on white to improve contrast.

12. The use of additional CRT's for alarm summaries and further alarm arrangement by priority is under study.

During October 6-10, 1980, the NRC performed a review/audit of the Watts Bar control room. Responses to NRC findings and proposed corrective actions were discussed with the NRC on May 7, 1981.

Note: The TVA responses will be provided to the NRC by letter and consolidated in this response.

The detail control room review (approximately one year in duration) will be conducted in accordance with NUREG-0700. However, we are continuing our own review and studies to identify changes that may be needed.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

NRC Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Changes to Previous Requirements and Guidance

There are no changes to previous guidance.

NRC Clarification

These requirements for the SPDS are being developed in NUREG-0696, which is scheduled for issuance in November 1980.

Implementation

Schedules for implementation will be issued in conjunction with issuance of NUREG-0696.

Type of Review

To be determined in conjunction with issuance of NUREG-0696.

Documentation Required

To be determined in conjunction with issuance of NUREG-0696.

Technical Specification Changes Required

To be determined in conjunction with issuance of NUREG-0696.

References

NUREG-0660, Item 1.D.2

NUREG-0696

PLANT SAFETY PARAMETER DISPLAY CONSOLE

RESPONSE

Information on the Safety Parameter Display System will be provided before fuel load.

I.G.1 TRAINING DURING LOW-POWER TESTING

NRC Position

The first special low power test program was developed by TVA, approved by the staff, and completed at the Sequoyah Nuclear Power Station. However, it was not the intention of NRC to obligate all succeeding applicants to the Sequoyah program. The purpose of this letter is to indicate our minimum requirements for an acceptable program of training during low power testing for PWR's, which will provide the basis for future test programs.

With respect to the subject TMI-related requirements, NUREG-0694 issued in June 1980, stated that applicants for new operating licenses will:

> Define and commit to a special low-power testing program approved by NRC 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 (Part 1, Paragraph I.G.1);
>
> Obtain nuclear steam supply system (NSSS) vendor review of low power testing procedures to further verify their adequacy. This requirement must be met before fuel loading (Part 1, Paragraph I.C.7); and
>
> Supplement operator training by completing the special low power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators. This requirement shall be met before issuance of a full-power license (Part 2, Paragraph I.G.1).

The above requirements are incorporated in Enclosure 2 to NUREG-0737, issued October 31, 1980, as 'Requirements Issued 6/26/80.'

NUREG-0694 also stated that for 'later operating license applcants' (those subsequent to the 'NTOL') 'the staff intends to conduct one operating license review by combining the fuel-loading and full-power testing requirements. Into a single set of operating license requrements.' Our safety evaluation report on your facility will include our complete evaluation of the special low power test program including details of the supplemental operator training your commit to perform.

Your program, as submitted for our review, sould 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 shoiuld 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 off site 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 be performed at those plants for which the test has not been demonstrated at a comparable prototype plant.

Our approval of your program will be subject to conformance with the above requirements and the following criteria:

The tests should not pose an undue risk to the public.

The risk of equipment damage must be low.

The format for your procedures should be consistent with Regulatory Guide 1.68. The procedures and your safety evaluation for your program should be submitted to I&E and NRR at least four weeks prior to the date of performing the tests.

The results of the special low power tests should be documented as part of the 'Startup Test Report' (see Regulatory Guide 1.16). Guide 1.68.

<u>Implementation</u>

The applicant should provide the required information four months before the scheduled Safety Evaluation Report.

<u>References</u>

NRC letter from R. L. Tedosco to H. G. Parris dated November 14, 1980.

TVA RESPONSE

I.G.1 Training During Low Power Testing

In accordance with attachment 5, TVA will perform one basic natural circulation test which will provide the required training objectives. The plant characteristics described in this letter have been demonstrated in the Sequoyah natural circulation test program and need not be repeated at Watts Bar. The proposed basic natural circulation test will be repeated several times to allow all available licensed reactor operators to participate in the initiation, maintenance, and recovery from natural circulation mode. The proposed test will essentially be a repetition of the basic natural circulation test conducted at Sequoyah. A description of the procedure will be included in FSAR Chapter 14 and a detailed procedure will be available for review before fuel loading on Watts Bar unit 1. We do not propose to repeat this on Watts Bar unit 2 since some of the operators will have received the unit 1 training in addition to the natural circulation training provided by the operator training program.

Attachment 5
UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 16 1981

Docket No(s).: 50-390
and 50-391

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr.

SUBJECT: TMI-2 ACTION PLAN ITEM I.G.1 - SPECIAL LOW POWER TESTING

NUREG-0737 "Clarification of TMI Action Plan Requirements" and NUREG-0694 "TMI Related Requirements for New Operating Licenses", Item I.G.1, calls for the implementation of "a special low power testing program approved by NRC 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". Some PWR applicants have committed to a series of natural circulation tests. To date such tests have been performed at the Sequoyah 1, North Anna 2, and Salem 2 facilities. Based on the success of the programs at these plants, the staff has concluded that augmented natural circulation training should be performed for all future PWR operating licenses. This is to be implemented by including descriptions of natural circulation tests in your FSAR (Chapter 14 - Initial Test Program). If they are not already included in your FSAR, the natural circulation tests and associated training should be included either by modifying existing or adding new test descriptions in accordance with Regulatory Guide 1.70, Paragraph 14.2.12. The tests should fulfill the following objectives:

Training

Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance and recovery from natural circulation mode. 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.

Testing

The 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 and without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.

If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the above training objectives.

Procedure Validation

The tests should make maximum practical use of written plant procedures to validate the completeness and accuracy of the procedures.

The natural circulation tests require a source of actual or simulated decay heat. The tests may be performed during initial startup using nuclear heat to simulate decay heat, or may be performed later in the initial fuel cycle when actual decay heat is adequate to permit meaningful testing. If the test objectives are not compromised, pump heat during forced circulation operation could provide an acceptable source of simulated decay heat (e.g., the Loss-of-Onsite and Offsite A/C Test performed at North Anna 2).

Applicants who perform a natural circulation boron-mixing and cooldown test to demonstrate compliance with Branch Technical Position RSB BTP 5-1 may use that test to accomplish some or all of the above training and testing objectives.

This guidance is provided for all PWR OL applicants and supersedes that dated November 14, 1980. Regulatory Guide 1.68 and/or the Standard Review Plan will be revised at a future date to include natural circulation testing and the associated training. OL applicants should submit test descriptions in accordance with Regulatory Guide 1.70, Paragraph 14.2.12, as part of their FSAR or an amendment thereto. Detailed test procedures should be made available for NRC review 60 days prior to scheduled test performance (see Regulatory Guide 1.68, Appendix B). When required by 10 CFR 50.59, a safety analysis must be prepared and distributed in accordance with the requirements stated therein.

Sincerely,

RLTedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

cc: See next page

II.B.1 REACTOR COOLANT SYSTEM VENTS

NRC Position

Each applicant 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 assures 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.

Changes to Previous Requirements and Guidance

(1) The probability of a valve failing to close, once opened, should be minimized.

(2) Establishes environmental qualification (Commission Order, May 23, 1980).

(3) Establishes provisions for testing.

(4) Delete requirements of September 27, 1979, letter from Vassallo to applicants stating that vents shall satisfy single-failure criteria of IEEE-279. Vent systems are not required to have redundant paths. A degree of redundancy should be provided by powering different vents from different emergency buses.

(5) Documentation date changed to July 1, 1981, and implementation date to July 1, 1982.

Clarification does not change NRC concept of requirement, but provides more detail on scope. The dates have been revised to provide time for procurement and installation.

*It was the intent of the October 30, 1979, letter to delete the requirement to meet the criteria of 10 CFR 50.44 and SRP 6.2.5 for beyond-design-basis events. The analysis requirements of Position 2 in the September 13, 1979, letter are therefore unnecessary.

NRC 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 PWR's, 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 BWR's, 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, and

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

C. 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 assures 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 assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

Implementation

Installation should take place by July 1, 1982. Until staff approval is obtained, installation may proceed; but operating procedures should not be implemented and valves should be placed in a condition so as to minimize the potential for inadvertent actuation (e.g., remove power).

Type of Review

A preimplementation review will be performed prior to authorizing use of the vent.

Documentation Required

By July 1, 1981, the licensee shall provide the following information on the reactor coolant vent system for staff review:

(1) The information requested in Items 1 and 2 under 'Position';

(2) A discussion of the design with respect to conformance to the design criteria discussed under 'Clarification,' including deviations, if any, with adequate justification for such deviations; and,

(3) Supporting information including logic diagrams, electrical schematics, piping and instrumentation diagrams, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Commission Orders, May 23, 1980 (CLI-80-21)

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

REACTOR COOLANT SYSTEM VENTS

TVA RESPONSE

The preliminary design of the Watts Bar vent system was provided to the NRC by letter dated May 8, 1980, from L. M. Mills to L. S. Rubenstein on the Sequoyah Nuclear Plant docket. TVA provided additional information on the vent design to the NRC by letter dated July 1, 1981 from M. R. Wisenburg to E. Adensam also on the Sequoyah Nuclear Plant docket.

The information submitted on the Sequoyah docket is directly applicable to Watts Bar and is summarized below.

The following discussion on venting of the reactor coolant system is divided into two separate areas: (1) a description of the reactor vessel head vent system and (2) a discussion on the additional pressurizer venting capability proposed for Watts Bar.

1. Reactor Vessel Head Vent System

   The basic function of the Reactor Vessel Head Vent System (RVHVS) is to remove noncondensable gases or steam from the reactor vessel head. This system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases in the RCS.

2. General Description

   The RVHVS is designed to remove noncondensable gases or steam from the RCS by remote manual operations from the control room. The system discharges either into the pressurizer relief tank or directly into a well-ventilated area of the containment. The RVHVS is designed to vent a volume of hydrogen at system design pressure and temperature approximately equivalent to one-half of the RCS volume in one hour.

   The RVHVS consists of two parallel flow paths with redundant isolation valves in each flow path. The venting operation uses only one of these flow paths at any one time.

   The active portion of the system consists of a series-parallel arrangement of four 1-inch solenoid-operated isolation valves connected to the upper head injection system piping, which is located on the reactor vessel head. The inboard solenoid-operated valves are open/close isolation valves. The outboard valves are remotely operated throttle valves. The system design with two valves in series after each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The isolation valves in one flow path are powered by one vital power supply and the valves in the second flow path are powered by a second vital power supply. The isolation valves are fail closed, normally closed active valves.

   If one single active failure prevents a venting operation through one flow path, the redundant path is available for venting. Similarly, the two isolation valves in each flow path provide a single failure method of isolating the venting system. With two valves in series, the failure of any one valve or power supply will not inadvertently open a vent path. Thus, the combination of safety-grade train assignments and valve failure modes will not prevent vessel head venting or venting isolation with any single active failure.

The RVHVS has two normally deenergized valves in series in each flow path. This arrangement eliminates the possibility of spuriousy opened flow path due to the spurious movement of one valve. As such, power lockout to any valve is not considered necessary.

The RVHVS connection to the upper head injection system includes a 3/8-inch orifice. This orifice forms the Safety Class 1 to Safety Class 2 transition. The system is orificed to limit the blowdown from a break downstream of the orifice to within the capacity of one centrifugal charging pump. From the orifice to the first anchor downstream of the second isolation valves, all piping and equipment are designed and fabricated in accordance with ASME Code, Section III, Class 2 requirements. Also, the vent piping from the downstream isolation valve to the containment is being designed and fabricated to ASME Section III, Class 2, requirements.

The inboard isolation valves for the RVHVS have stem position switches. The position indication from each valve is monitored in the control room by status lights. The outboard throttle (isolation) valves for the RVHVS are equipped with Linear Variable Differential Transformers (LVDT) from which a positive valve position indication is provided as a feedback signal to the valve controller and then on to the control room. Thus, the requirement for valve position indication is satisfied.

The four solenoid-operated isolation valves (two inboard open/close isolation valves and two outboard throttle valves) for the RVHVS will be qualified to IEEE-323-1974, -344-1975, and -382-1972. These isolation valves are fail closed active valves. These isolation valves are also included in the Westinghouse valve operability program which is an acceptable alternative to Regulatory Guide 1.48. The associated environmental and seismic qualification of (1) the positive valve position indication system/devices (i.e., the stem position switches for the inboard isolation valves and the LVDT's for the outboard throttle valves), and (2) the control components for the throttle valves (i.e., the valve controller) will be performed in conjunction with the qualification of the valves.

The RVHVS can be tested for operability by cycling each valve through one full cycle from the control room. Valve movement can be verified either locally or by observing the status lights of other position indication devices. In keeping with testing frequencies of valves in safeguard and related systems, a surveillance interval of 18 months is being considered. If it is desired to demonstrate that a flow path is opened, this could be accomplished during normal RCS venting following refueling outages. Detailed test specifications and procedures are not as yet available.

Venting of the pressurizer is currently provided in the Watts Bar design by the pressurizer relief valves. However, TVA is presently negotiating with Westinghouse to add a pressurizer vent system to the present RCS vent design. The system design basis, piping and valve classifications and qualification, power supply, and position indication requirements would be essentially the same as for the RVHVS. It is anticipated that the pressurizer vent system would interface with the RVHVS downstream of the RVHVS isolation valves, thereby allowing both systems to utilize common venting areas inside containment.

Additional information will be provided as the overall RCS vent design progresses.

TVA will install the head vent system before fuel load at Watts Bar.

The venting guidelines being developed by the Westinghouse Owners' Group will be incorporated into the Watts Bar Emergency Operating Instructions. This guideline will be provided to the NRC for review by the Owners' Group.

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS

NRC Position

With the assumption of a postaccident release or radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% 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.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters to all operating nuclear power plants, dated September 13 and October 30, 1979, and was incorporated into NUREG-0660. Significant changes in requirements or guidance are:

(1) Adds several areas to be evaluated for access to ensure that these areas are not overlooked.

(2) Specifies that the source term for recirculated depressurized coolant need not be assumed to contain noble gas since this gas will be released from the liquid when it is depressurized.

(3) Specifies that certain systems be considered as potential sources and that leakage from systems outside of containment need not be considered as potential sources.

(4) Allows averaging over 30 days of the dose rate criteria for areas requiring continuous occupancy and that the control room and technical support center should be considered areas requiring continuous occupancy. This ensures that the dose rate criteria is applied correctly to these areas.

(5) Specifies source terms to be used in conjunction with Commission Order and Memorandum dated May 23, 1980, (CLI-80-21) on equipment qualification, and specifies schedule in above order.

(6) Because of difficulty in obtaining equipment (e.g.,

remote-operated valves), the implementation date is moved to January 1, 1982, or the first outage of sufficient duration thereafter, but no later than July 1, 1982.

NRC 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 assure that personnel can perform 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, 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., you may assume the radioactive decay that occurs before fission products can be transported to various systems).

(a) Liquid-Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% 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% of the core equilibrium noble gas inventory and 25% 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 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 leakage of systems located outside of 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 <u>not</u> 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% noble gases, 50% iodines, and 1% particulates) to the containment atmosphere.

(b) LOCA events in which the primary system may not depressurize should consider the source term (100% noble gases, 50% iodines, and 1% 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 containment. Non-LOCA events both inside and outside of containment should use 10% noble gases, 10% iodines, and 0% 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

Implementation

(1) For Vital Area Access

By January 1, 1982, modifications should be completed: For OL applicants, documentation of the evaluation should be completed at least four months before the operating license is issued.

(2) For Equipment Qualification

All safety-related electrical equipment must be fully qualified by June 30, 1982. Documentation in accordance with:

Operating Licenses (operating license expected by June 30, 1982): submittal no later than 4 months before issuance of operating license. Operating licenses in accordance with review schedule.

Type of Review

A postimplimentation review will be performed.

Documentation Required

For Vital Area Access--For operating license applicants provide a summary of the shielding design review, a description of the results of this review, and a description of the modifications made or to be made to implement the result of the review. Include in your submittal:

(1) Specification of source terms used in the evaluation; including time after shtudown that was assumed for source terms in systems;

(2) Specification of systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation. If any of the systems listed in 'Clarification,' item 2, were excluded, explain why such systems are excluded from review;

(3) Specification of areas where access is considered necessary for vital system operation after an accident. If any of the areas listed in the 'Clarification' section above were not considered to be areas requiring access after an accident, explain why they were excluded;

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

Technical Specification Changes Required

Technical specifications will not be required.

References

NUREG-0578, Recommendation 2.1.6.b

NUREG-0660, Item II.B.2

Commission Order and Memorandum, May 23, 1980 (CLI-80-21)

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated April 25, 1980.

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, dated May 7, 1980.

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS

TVA RESPONSE

The Watts Bar design bases include the assumption of TID 14844 sources. TVA plants are specifically designed to mitigate major design basis events with no access outside the main control room (MCR) being required. With this goal in mind, the plants were not specifically designed for any access outside the MCR. To specifically design for guaranteed access at any time in most parts of the auxiliary building is not feasible. However, the current designs allow considerable capability for access for short times if the entry time into the area can be selectively chosen.

The current arrangements and shielding for normal operation will help minimize the impact from post-accident contained sources even though the shielding was not intended for that purpose. In certain instances, TVA has provided some shielding for post-accident access. TVA has performed a shielding review for Watts Bar. The review included generation of radiation source terms for primary system water and containment sump water based on TID 14844. These fluids were assumed to circulate in the plant systems designed for accident response and also in systems used in normal plant operation but which might be called upon for accident recovery. From the analyses performed, radiation doses can be determined at locations in the plant near accident recovery equipment.

Watts Bar is designed to mitigate major accidents without access to the plant outside the MCR. Two areas outside the MCR were identified which would be helpful in responding to an accident situation. One area is a control panel in the shutdown board area at elevation 734.0. The panel is immediately outside the MCR. There are no contained sources in this area and direct gamma doses will not cause any concern for access. The other area is the normal plant sampling station in the auxiliary building at elevation 690.0. Dose rates in the sample room were evaluated for various times into the accident. a representative value at one hour into the accident is 900 mr/minute. Sampling procedures for accident situations in the interim period until a redesigned sampling facility can be installed take into account these calculated values. If samples are ever needed in an accident, the procedure will also utilize actual dose rate measurements to evaluate accessibility and occupancy times.

As a result of this study, it has been determined that no additional shielding is necessary at Watts Bar, except for lead blankets around sample lines in the sample room to improve its accessibility in an accident situation.

A detailed report on TVA's shielding review for Sequoyah was transmitted to A. Schwencer by L. M. Mills letter dated June 16, 1980. This information is also applicable to Watts Bar.

Watts Bar Nuclear Plant will meet the requirements of GDC 19.

NRC 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 Ci/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 shileding 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 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 postaccident 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 absorbers and high-efficiency particulate air (HEPA) filters.

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

<u>Implementation</u>

Installation should take place by January 1, 1982.

<u>Type of Review</u>

A postimplementation review will be performed.

<u>Documentation Required</u>

Operating License Applicants--Provide a description of the implementation of the position and clarification including P&IDs, together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample analysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis, in accordance with the proposed review schedule but in no case less than 4 months prior to the issuance of an operating license.

<u>Technical Specification Changes Required</u>

Changes to technical specifications will be required.

<u>References</u>

NUREG-0578, Recommendation 2.1.8.a

NUREG-0660, Item II.B.3

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

ATTACHMENT 15

POST ACCIDENT SAMPLING CAPABILITY

TVA RESPONSE

A design and operational review of the reactor coolant sampling systems and analysis facilities has been performed. TVA expects to complete required modifications in accordance with the schedules established by NRC provided equipment procurement/installation conflicts are not encountered. These modifications will make provisions for sampling water from the reactor coolant system and the containment for the degraded accident condition.

To enhance the capability at Watts Bar for post-LOCA sampling, TVA will:

1. Make provisions for sampling water from the reactor coolant system and the containment sump for the degraded accident condition.

2. Install new lines with connections to the existing gaseous radiation sampling system for use in sampling the containment atmosphere for the degraded accident conditions.

3. Route sample lines to a shielded sampling station in an accessible area and provide for taking samples which could be removed offsite for analysis.

TVA will provide the capability to obtain and analyze (within three hours) reactor coolant samples and containment air samples under accident conditions.

TVA will provide, as practical, onsite radiological and chemical analysis capabilities in order to quantify the following:

1. core damage (RCS)

2. hydrogen level in containment

3. dissolved gases and boron concentration (RCS)

Provisions will be made:

1. to permit sampling under both positive and negative pressure,

2. for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion for preventing blockage of sample lines, for appropriate disposal of samples and for passive flow restrictions,

3. and to qualify the sampling system to appropriate seismic and environmental requirements.

The radiological sample analysis capability will include provisions to:

1. Identify and quantify isotopes to levels corresponding to the source terms. The ability to dilute samples and to measure nuclide concentrations as low as 1 Ci/gm will be provided.

2. Restrict background levels in the laboratory to meet NUREG requirement for personnel dose.

3. Maintain plant procedures to identify the analysis required, measurement techniques and provisions for reducing background.

The chemical analysis capability will consider the presence of the radiological source term indicated by the radiological analysis.

Procedural changes and plant modifications will be met to ensure that radiation exposures are within NUREG requirements.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

NRC Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Changes to Previous Requirements and Guidance

Persons who must participate in the training program are to be defined.

The implementation schedule has been revised to reflect the TMI Action Plan schedule.

NRC Clarification

Shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators shall receive all the training indicated in Enclosure 3 to H. R. Denton's March 28, 1980, letter.

Managers and technicians in the Instrumentation and Control (I&C), health physics, and chemistry departments shall receive training commensurate with their responsibilities.

Implementation

Applicants for operating licenses should develop a training program prior to fuel loading and complete the program prior to full-power operation.

Type of Review

A postimplementation review will be performed.

Documentation Required

Programs shall be available for review.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item II.B

Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, dated March 28, 1980.

TVA RESPONSE

II.B.4 Training For Mitigating Core Damage

TVA has completed development and implementation of the training program for the shift technical advisors and operations personnel from the operations supervisor to the licensed operators to comply with enclosure 3 of H. R. Denton's March 28, 1980 letter. An abbreviated program of the operator training will be presented to managers and technicians in the Health Physics, Plant Chemistry, and Instrumentation and Controls Sections commensurate with their responsibilities in the event of a core damaging accident. The initial training program for Watts Bar will be completed by fuel loading.

II.D.1 PERFORMANCE TESTING OR BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

NRC Position

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

Changes to Previous Requirements and Guidance

A. Safety and Relief Valves and Piping--The types of documentation required for safety and relief valves and piping and the specific submittal dates are considered to be a clarification of item II.D.1 as described in NUREG-0660. The submittal of information was implied but not explicitly discussed in that report.

B. Block Valves--Qualification of PWR block valves is a new requirement. Since block valves must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating a small loss-of-coolant accident due to a stuck-open relief valve. Isolation of a stuck-open power-operated relief valve (PORV) is not required to ensure safe plant shutdown. However isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core-cooling system. Repeated unnecessary challenges to these system are undersirable.

C. ATWS Testing--Testing of anticipated transients without scram (ATWS) for later phases of the valve qualification program was noted in Item II.D.1 of NUREG-0660. The clarification below provides updated information on PWR ATWS temperature and pressure conditions and clarifies that ATWS testing need not be accomplished by July 1981.

NRC Clarification

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:

(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 or 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 3200 to 3500 (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.

Implementation

See implementation schedules in the 'Documentation Required' section.

Type of Review

Preimplementation review will be performed for EPRI test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed.

Post implementation review will also be performed of the test data and test results as applied to plant-specific situations.

Documentation Required

Preimplementation review will be based on EPRI and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980
Final BWR Test Program--October 1, 1980
Block Valve Qualification Program--January 1, 1981

Postimplementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI) Generic Test Program Results--July 1, 1981
Plant-specific submittals confirming adequacy of safety and relief valves based on applicant preliminary review of generic test program results--July 1, 1981
Plant-specific reports for safety and relief valve qualification--October 1, 1981
Plant-specific submittals for piping and support evaluations--January 1, 1982
Plant-specific submittals for block valve qualification--July 1, 1982

Technical Specification Changes Required

No technical specification changes are required.

References

NUREG-0578

NUREG-0660, Item II.D.1

PERFORMANCE TESTING OF PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES

TVA RESPONSE

TVA is a participant in the EPRI PWR Relief and Safety Valve Test Program. Submittal dates will be updated by EPRI as the test program progress.

II.D.3 DIRECT INDICATION OF RELIEF-AND SAFETY-VALVE POSITION

NRC 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.

Changes to Previous Requirements and Guidance

There is no changes to the previous requirements.

NRC 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, and

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

Implementation

Implementation will be completed prior to the issuance of operating license.

Type of Review

A preimplementation review will be performed.

Documentation Required

Documentation should be provided that discusses each item of the clarification, as well as electrical schematics and proposed test procedures in accordance with the proposed review schedule, but in no case less than 4 months prior to the scheduled issuance of the staff safety evaluation report.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.a

NUREG-0660, Item II.D.3

NUREG-0694, Part 1

Commission Order and Memorandum, May 23, 1980 (CLI-20-81)

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated September 27, 1979.

Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, dated November 9, 1979.

DIRECT INDICATION OF RELIEF-AND-SAFETY-VALVE POSITION

TVA RESPONSE

The power operated relief valves have a reliable direct, stem-mounted position indication in the main control room. Valve position of the pressurizer safety valves is currently provided in the following manner.

1. Temperature is sensed downstream of the valves and displayed in the main control room including high temperature alarms.

2. The pressurizer relief tank has temperature, pressure, and fluid level indication and alarms in the main control room.

3. The pressurizer has high pressure alarms in the main control room.

An acoustic monitoring system for the three safety relief valves and Power-operated Relief Valves has been provided on unit 1 and will be provided on unit 2 before fuel loading. An accelerometer is mounted on the valve discharge line just downstream of each valve. The accelerometer signals go to a charge converter inside containment mounted in a NEMA-4 enclosure. A valve flow indicator module is located in the main control room. The flow indicator module gives positive indication of the fully open and fully closed position of each valve. An alarm in the main control room will indicate when any valve is not in the fully closed position.

This design provides the operator with unambiguous indication of valve position as specified in the above response.

Valve position is indicated in the main control room and alarmed as discussed in the above response.

Valve position indication for Watts Bar Nuclear Plant meets seismic and environmental qualification requirements as specified for Sequoyah. Technology for Energy Corporation (TEC), the vendor for the monitoring system is currently conducting a qualified life test program.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

NRC Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR 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 for Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

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

Changes to Previous Requirements and Guidance

The date for implementation of short-term requirements has been slipped because staff review of submittals is not complete.

NRC Clarification

Operating License Applicants--Operating license applicants have been requested to respond to the staff letter of March 10, 1980. These responses will be reviewed during the normal review process for these applications.

Implementation

Applicants for operating license should refer to the letters of March 10, 1980.

Type of Review

A preimplementation review will be performed.

Documentation Required

Applicants will be required to submit the information indicated above.

Technical Specification Changes Required

Changes to technical specifications will be determined by specific item.

Reference

NUREG-0660, Item II.E.1.1

Letter from D. F. Ross, Jr., NRC, to All Pending W and C-E License Applicants, dated March 10, 1980.

Letter from D. F. Ross, Jr., NRC, to All Pending B&W License Applicants, dated April 24, 1980.

TVA RESPONSE

The Reliability and Availability Section has compared the WBN auxiliary feedwater system (AFWS) to the SQN AFWS. We believe the analysis done on the SQN AFWS is applicable to the WBN AFWS with two exceptions:

(1) At WBN only one condensate storage tank (CST) is aligned to provide water immediately to the AFWS for each unit. The availability of this source of water is high and, if necessary, the other CST could be used as a water source by opening the two manual valves (2-502 and 2-503) which connect the outlet of the CST's. This WBN design has the advantage over the SQN design of separate piping to the AFW pumps for each unit; therefore, a failure of one line to supply water to the AFW pumps does not affect the performance of the AFWS for the other unit.

(2) At Watts Bar two check valves are provided between the containment penetration and the AFW inlet to SG 1 and 4. An additional check valve is located at the inlet of SG 2 and 3. The SQN AFWS design includes two check valves between containment and the SG inlet of SG 2 and 3. At Sequoyah no check valves are installed in this area for SG 1 and 4. The additional check valve on SG 2 and 3 and the two check valves on SG 1 and 4 at Watts Bar add a very small amount of probability of failing to deliver water to the SG. However, they provide increased protection against reverse flow from the steam generator in the case of a line break.

There are several other differences in the design and equipment of the WBN AFWS. However, these differences have no effect on the level of redundancy in the system and the minimum amount of equipment which would be necessary for the WBN AFWS to fulfill its safety function. Also, the differences in the design and equipment are so small that we believe it would not be necessary to modify the input reliability data.

This comparison has shown the WBN AFWS to be essentially equal in reliability to the SQN AFWS with some extra protection against piping ruptures.

II.E.1.2 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

PART 1: Auixiliary Feedwater System Automatic Initiation

NRC Position

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 (AFWS), the following requirements shall be implemented in the short term:

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

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

(3) Testability of the initiating 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 AFWS 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 AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

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

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

Changes to Previous Requirements and Guidance

There are no changes to the previous guidance issued in the H. R. Denton letter to licensees, dated October 30, 1979.

NRC Clarification

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

The staff has determined that the following salient paragraphs of IEEE 279-1971 should be addressed as a minimum:

IEEE 279-1971, Paragraph

4.1*	General Functional Requirements
4.2*	Single Failure
4.3, & 4.4	Qualification
4.6	Channel Independence
4.7	Control and Protection System Interaction
4.9* & 4.10*	Capability for Testing
4.11	Channel Bypass
4.12	Operating Bypass
4.13	Indication of Bypass
4.17*	Manual Initiation

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that the above requirements are met. The documentation should include as a minimum:

(1) A discussion of the design with respect to the above paragraphs of IEEE 279-1971; and

(2) Supporting information including system design description, logic diagrams, electrical schematics, piping and instrument diagrams, test procedures, and technical specifications.

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*These requirements were part of the short-term, control-grade requirements.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.7.a

NUREG-0660, Item II.E.1.2

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

PART 2: Auxiliary Feedwater System Flowrate Indication

NRC Position

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 AFWS 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.

Changes to Previous Requirements and Guidance

The requirements have been relaxed to require only a single-channel flow indication, instead of redundant channels. This single channel need not be seismically qualified nor need it be powered from a Class IE power source.

The auxiliary feedwater flow indication requirements have been relaxed because flow indication is of secondary importance in assuring steam generator cooling capability for Westinghouse steam generators.

NRC Clarification

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

1. 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 IE power source.
   (c) periodically testable
   (d) part of plant quality assurance program
   (e) capable of display on demand

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, and

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

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that the above-specified requirements have been met. The documentation should include as a minimum:

(1) A discussion of the design with respect to each of the requirements specified above; and

(2) Supporting information including system design description, logic diagrams, electrical schematics, piping and instrument diagrams, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.7.b

NUREG-0660, Item II.E.1.2

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

TVA RESPONSE

A. AUTOMATIC INITIATION

The auxiliary feedwater system is automatically initiated by redundant, coincident logic to preclude loss of function due to a single failure and to provide on line testability. The auxiliary feedwater system and initiating logic are described in FSAR Section 10.4.9. The auxiliary feedwater control circuitry including the automatic initiating circuitry is safety-grade, Class IE, and is powered from a power source connected to the emergency power system. Each auxiliary feedwater pump has manual initiation capability independent of the automatic initiation. The ac motor-driven pumps and valves are included in the automatic alignment of the loads to the emergency power system.

Features of the Watts Bar AFW System include:

1. Automatic and manual initiation of AFW are provided.

2. On line testability is provided.

3. Initiating signals are powered from the emergency power system.

4. The ac motor-driven pumps and valves are included in the automatic alignment of loads to the emergency power system.

5. Manual initiation capability is provided independent of the automatic initiation.

6. Appropriate electric power is supplied via the emergency power system for all valves where control air is needed for operation.

B. FLOW INDICATION

The auxiliary feedwater flow indication at Watts Bar consists of redundant, diverse, safety-grade transmitter loops for each steam generator. The redundant transmitters are mounted in separate seismically qualified panels and are powered from separate power sources connected to the emergency power system. The cables for the redundant transmitter loops are routed in separate low level signal trays which are kept separate from all power cables. The auxiliary feedwater flow instrument channels are powered from emergency buses (120v ac vital instrument buses) consistent with the separation requirements for redundancy and the diversity requirements for the AFW's.

II.E.3.1 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

NRC 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements in October 30, 1979 letter from H. R. Denton to all licensees.

NRC Clarification

(1) Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only one Class IE 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; and

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

(6) The Class IE 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 IE 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).

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license.

Type of Review

A review will be performed as part of the licensing review process.

Documentation Required

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that each of the subparts of the position stated above are met. The documentation should include as a minimum, supporting information including system design description, logic diagrams, electrical schematics, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.1

NUREG-0660, Item II.E.3.1

NUREG-0694, Part 2

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

TVA RESPONSE

The pressurizer heaters are powered and controlled from Class IE sources (see FSAR Figures 8.3-16, 8.3-17, 8.3-18, and 8.3-19). The motive and control power interfaces with the emergency buses are qualified in accordance with safety-grade requirements. All four heater banks will trip on a safety injection signal when in the normal mode. After safety injection reset and level recovery in the pressurizer, one backup heater bank (1C) would operate automatically. The other two backup heater banks and the control bank would not come on automatically but are manually activated. In the event of a loss of offsite power and safety injection signal, two backup heater banks rated at 485 KW each can be manually activated by hand switches in the main control room, 90 seconds after emergency power becomes available. The required operator actions are specified in the Watts Bar Emergency Operating Instructions.

As specified, the Watts Bar design provides redundant capability for providing emergency power to each bank of heaters. The independence of the Class IE division power supply for each heater bank is shown by the following load group designation.

Power Train	Heater Bank
1A-A	1A-A, 1D
1B-B	1B-B, 1C
2A-A	2A-A, 2D
2B-B	2B-B

The pressurizer heaters are automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal (SIS). SIS reset is covered in the Watts Bar EOI.

Emergency power is available to heaters required for maintaining natural circulation in a hot standby condition.

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

NRC Position

Plants using external recombiners or purge systems for postaccident 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.

Changes to Previous Requirements and Guidance

Changes in the implementation date have been made because of equipment procurement problems and to minimize the number of plant shutdowns necessary must make to install equipment related to the TMI Action Plan.

NRC 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 for 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 now 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.

Implementation

Operating license applicants must have design changes completed prior to issuance of an operating license.

Documentation Required

The licensees shall inform the NRC when the required design modifications have been completed.

Technical Specification Changes Required

Changes to technical specifications will be required for plants that need to make modifications.

References

NUREG-0578

Letter from H. R. Denton, NRC, to All Operating Reactor Plants, dated October 30, 1979.

DEDICATED HYDROGEN PENETRATIONS

TVA RESPONSE

Watts Bar Nuclear Plant does not use external recombiners or purge systems for post accident combustible gas control.

The Watts Bar design utilizes a manually actuated ESF recombiner system inside containment which is redundant and fully qualified. (See FSAR Section 6.2.5)

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

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 reevaluation 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 is enclosed as Attachment 1.)

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

Changes to Previous Requirements and Guidance

Although there has been no change in the requirements since NUREG-0660 was issued, positions 5, 6, and 7 have been previously transmitted to licensees. These three positions were not part of the original NUREG-0578 requirements of Recommendation 2.1.4; however they were added to item II.E.4.1 of NUREG-0660 as a result of further staff evaluation of features needed to improve containment isolation dependability.

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 assure 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.

Implementation

Applicants for an operating license must be in compliance with all positions before receiving an operating license.

Applicants must provide, and justify, the minimum containment pressure that will be used for initiating containment isolation as stated in position 5.

Type of Review

A postimplementation review will be performed for operating reactors.

Documentation Required

The type and dates of documentation required are as previously stated.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.4

NUREG-0660, Item II.E.4.2

Standard Review Plan, Section 6.2.4

II.E.4.2 ATTACHMENT 1, OCTOBER 23, 1979* INTERIM POSITION FOR CONTAINMENT PURGE AND VENT VALVE OPERATION PENDING RESOLUTION OF ISOLATION VALVE OPERABILITY

Once the conditions listed below are met, restrictons on use of the containment purge and vent system isolation valves will be revised based on our review of your responses to the November 1978 letter on this subject justifying your proposed operational mode. The November 1978 letters to all licensees identified certain events related to containment purging of concern to the NRC and requested commitments to either cease purging or justify purging operations. The revised restrictions can be established separately for each system.

(1) Whenever the containment integrity is required, emphasis should be placed on operating the containment in a passive mode as much as possible and on limiting all purging and venting times to as low as achievable. To justify venting or purging, there must be an established need to improve working conditions to perform a safety-related surveillance or safety-related maintenance procedure. (Examples of improved working conditions would include deinerting, reducing temprature,** humidity, and airborne activity sufficiently to permit efficient performance or to significantly reduce occupational radiation exposures.)

(2) Maintain the containment purge and vent isolation valves closed whenever the reactor is not in the cold shutdown or refueling mode until such time as you can show that:

(a) All isolation valves greater than 3-in. nominal diameter used for containment purge and venting operations are operable under the most severe design-basis-accident (DBA) flow-condition loading and can close within the time limit stated in the technical specifications, design criteria, or operating procedures. The operability of butterfly valves may, on an interim basis, be demonstrated by limiting the valve to be no more than 30° to 50° open (90° being full open). The maximum opening shall be determined in consultation with the valve supplier. The valve opening must be such that the critical valve parts will not be damaged by DBA-LOCA (loss-of-coolant accident) loads and that the valve will end to close when the fluid dynamic forces are introduced, and

(b) Modifications, as necessary, have been made to segregate the containment ventilation isolation signals to ensure that, as a minimum, at least one of the automatic safety injection actuation signals is uninhibited and operable to initiate valve closure when any other isolation signal may be blocked, reset, or overridden.

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*Previously referred to as DOE Interim Position.
**Only when temperature and humidity controls are not in the present design.

CONTAINMENT ISOLATION DEPENDABILITY

TVA RESPONSE

Watts Bar Nuclear (WBN) Plant meets all of the NRC positions concerning containment isolation. Specific information pertaining to each of the positions is given below.

1. The containment isolation system is designed to operate in two stages: Phase A and Phase B. Phase A isolates all process lines except safety injection, containment spray, portions of component cooling water, essential raw cooling water, and control air. Phase B isolates all remaining process lines except safety injection, containment spray, and auxiliary feedwater. The containment isolation design utilizes the concept of diversity of initiating signals. Phase A isolation can be initiated manually and is initiated by automatic or manual safety injection (SI) actuation. The SI signal is derived from (1) high steam line flow concident with low steam line pressure or low-low average reactor coolant average temperature, (2) high steam line differential pressure between loops, (3) low pressurizer pressure, or (4) high containment pressure. Phase B isolation can be initiated manually or automatically on a high-high containment pressure signal. The high-high containment pressure signal is redundant, Class IE circuitry. In addition, isolation valves in the primary containment ventilation system actuate on manual initiation of Phase A, Phase B, or SI and automatically on SI or high radiation signals.

2. A study was undertaken by TVA to (a) examine each system which penetrates the containment, (b) determine whether or not it is essential, (c) describe basis for this determination, (d) modify design if required.

   Every system that penetrates containment has been reevaluated to determine if it should be classified as essential or nonessential. The current classifications have been found to be acceptable and no changes in classification are planned.

3. The WBN design complies with NRC requirements on the automatic isolation of nonessential systems.

4. The WBN design complies with the NRC's requirements by requiring manual actions on the controls of individual components should it be necessary to change their status after the containment isolation signal has been cleared.

5. Qualified diverse containment isolation signals are provided.

   The containment isolation system is designed to prevent the release of radioactive material to the environment after an accident while ensuring that systems important for post accident mitigation are operational.

   Isolation is provided on the following three levels:

1. Nonessential systems - These systems are not required for post accident mitigation and are isolated automatically upon receipt of a Phase A isolation signal.

2. Essential systems - This group consists of the emergency core cooling systems, the containment spray system, and post accident H_2 monitors. These systems are not automatically isolated in the event of an accident. Remote manual valves are provided to permit isolation of these lines from the main control room if necessary.

3. Desirable systems - Systems that, while not required, significantly increase the plant's ability to cope with a small steam line break or LOCA. The systems are isolated automatically upon the receipt of a Phase B isolation signal. The systems falling into this category are emergency raw cooling water to the reactor coolant pumps (RCP) and containment coolers, component cooling water to the RCP's and control air.

Each line penetrating primary containment has been reviewed to ensure that (1) isolation of the line was based on its need to be in service post accident and (2) that each containment isolation valve received the proper isolation signal.

The containment isolation system is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. Before reset of the initiating signal, a valve closed by the signal can be opened only by constant operator demand with a valve's individual hand switch. The valve will return to the containment isolation position when the operator releases the hand switch. Resetting the isolation signal will not cause a containment isolation valve to change position. Each valve must be individually operated to cause a change from the containment isolation position.

The isolation of ventilation lines and lines that carry potentially radioactive fluid outside containment during power operation received special consideration at WBN. The ventilation lines receive high radiation signals in addition to the Phase A or B isolation signals. At present, the isolation of fluid lines that carry potentially radioactive material outside containment occurs upon the receipt of Phase A signal. This isolation signal would preclude the type of releases of radioactive material that occurred at TMI.

II.F.1 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION

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 postaccident 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; and

(6) Containment hydrogen concentration monitor.

NUREG-0578 provided the basic requirements associated with items (1) thourgh (3) above. Letters 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 NRC 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-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, and

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

References

NUREG-0660, Item II.F.1

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.1, ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR

NRC Position

Noble gas effluent monitors shall be isntalled 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 condition (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 ten.

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters dated September 13 and October 30, 1979. Significant changes in requirements or guidance are:

(1) Deletion of specific range overlap requirement.

(2) Specifies that offline monitoring is not required for safety valve and dump valve discharge lines.

(3) Implementation date changed from January 1, 1981 to January 1, 1982.

(4) Specifies that inline sensors are acceptable for concentrations between 10 2#μCi/cc to 10 5#μCi/cc of noble gas.

NRC Clarification

(1) Provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in Table II.F.1-1. 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 and accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gas.

(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 assure 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; and,

(v) the source of power to be used.

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

Implementation

Implementation must be completed by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

Licensing applicants should have available for review the final deisgn description of the as-built system, including piping and instrument diagrams together with either (1) a description of procedures for system operation and calibration, or (2) copies of procedures for system operation and calibration.

License applicants will submit the above details no less than 4 months prior to the issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

American National Standard ANSI N13.1-1969, February 1969

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979

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 by 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 detector (Note: 1R/hr 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, PWR drywell purge through the standby gas treatment system).

– Undiluted PWR condenser air removal system exhaust.

10 4#µCi/cc – Diluted containment exhaust gased (e.g., > 10:1 dilution,as with auxiliary building exhaust air).

– BWR reactor building (secondary containment) exhaust air.

– PWR secondary containment exhaust air.

10 5#µCi/cc – Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).

– PWR steam safety valve discharge, atmospheric steam dump valve discharge.

10 2#µCi/cc – Other release points (e.g., radwaste buildings, 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 data 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 of 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.

Upsteam filtration (perfiltering to remove radioactive idodines 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 in account in accounting for low-energy gamma radiation.

II.F.1, ATTACHMENT 2, SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

NRC Position

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

Changes to Previous Requirements and Guidance

This requirement was originally issued by letters dated September 13, 1979 and October 30, 1979. This requirement was inadvertently omitted from NUREG-0660. Significant changes in requirements or guidance are:

(1) Changes implementation date to January 1, 1982.

(2) Specifies a shiedling basis design envelope for design of samplers and sample transport devices.

(3) Specifies provisions for isokinetic sampling.

(4) Specifies representative sampling per criteria of ANSI N131-1969.

(5) Allows use of gamma radiation measurement and shielding/distance factors in lieu of analysis of highly radioactive samples.

NRC Clarification

(1) Provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of Table II.F.1-2. 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. 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 cirteria 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%. Further departure from the isokinetic condition need not be considered in design. Corrections for non-isokinetic 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.

Implementation

This requirement will be implemented by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

License applicants will submit the above details no less than 4 months prior to the issuance of an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.8.b

American National Standard ANSI N13.1-1969, February 1969

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

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 energey (E) of 0.5 MeV.

SAMPLING MEDIA

- Iodine > 90% effective adsorption for all forms of gaseous iodine.
- Particulates > 90% effective retention for 0.3 micron (μ) 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.

(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 as to provide a reasonable assessment of area radiation conditions inside 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 an should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended becasuse of potential maintenance difficulties.

(4) 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. 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.

Implementation Date

License applicants will submit the required documentation in accordance with the appropriate review schedule, but is no case len than four months prior to the issuance of the staff evaluation report for an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

For operating licenses applicants, provide a descritpion of the installed high-range containment monitors and specify the locations of these monitors inside containment. The descritpion of the monitors should include:

(1) The description of or name of manufacturer and model number of the monitors;

(2) Verification that the monitors meet the specifications of Table II.F.1-3;

(3) Verification that the monitors will be operable on January 1, 1982; and,

(4) A plant layout drawing showing the location of the monitors.

Technical Specification Changes Required

Chagnes to technical specifications will be required.

<u>References</u>

NUREG-0578, Recommendation 2.1.8.b

NUREG-0660

Regulatory Gudie 1.97, Revision 2

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

TABLE II.F.1-3 CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	–	The capability to detect and measure the radiation level witin 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 A, except as listed below.
SPECIAL CALIBRATION	–	In situ calibration by elkectronic 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 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.

II.F.1, ATTACHMENT 4, CONTIANMENT PRESSURE MONITOR

NRC 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.

Changes to Previous Requirements and Guidance

Regulatory Guide 1.97, Rev. 2 has been referenced since the October 30, 1979 letter as the guide for the design and qualification criteria for the containment pressure monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 has been added to this letter, Appendix A, and this is to be considered a new requirement.

The implementation date has been changed because of the new requirements and because of equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

NRC Clarification

(1) Design and qualification criteria are outlined in Appendix A.

(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.

Implementation

Operating license applicants with an operating license dated before January 1, 1982 must have design changes completed by January 1, 1982; those applicants with license dated after January 1, 1982 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensees shall inform the NRC when the required design modificatons have been completed. Applicants with operating license dates beyond January 1, 1982 shall provide the required design information at least six months before the expected date of operation.

Technical Specification Changes Required

Changes to technical specification will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.1, ATTACHMENT 5, CONTAINMENT WATER LEVEL MONITOR

NRC 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 PWR's and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity.

Change to Previous Requirements and Guidance

Regulatory Guide 1.97, Rev. 2 was referenced in the October 30, 1979, letter as the guide for the design and qualification criteria for the wide range containment water level monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 has been added to this letter (Appendix A) and this is to be considered a new requirement.

The implementation date has been changed because of the new requirements and because of equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

NRC Clarification

(1) The containment wide-range water level indication channels shall meet the design and qualification cirteria as outlined in Appendix A. 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 requirements 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) The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

Implementation

Applicants with operating license dates past July 1, 1981 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplementation review will be performed.

A preimplementation review will be performed for applicants for an operating license after January 1, 1982.

Documentation Required

Submittals applicants for operating licenses (with an operating license date before January 1, 1982) shall be provided by January 1, 1982. Applicants with operating license dates beyond January 1, 1982, shall provide the required design information at least six months before the expected date of operation.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.1, ATTACHMENT 6, CONTAINMENT HYDROGEN MONITOR

NRC 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% hydrogen concentration under both positive and negative ambient pressure.

Changes to Previous Requirements and Guidance

Regulatory Guide 1.97, Rev. 2 was referenced in the October 30, 1979 letter as the guide for the design and qualification criteria for the containment hydrogen monitor. However, there have been many changes made to this proposed revision and it has not yet been made final. Therefore, the appropriate sections of the latest version of Regulatory Guide 1.97 have been added to this letter (Appendix A) and, therefore, this is to be considered a new requirement.

The implementation date has been changed due to equipment procurement problems. The new implementation schedule is intended to allow licensees enough time to complete design modifications with a minimum number of plant shutdowns.

NRC Clarification

(1) Design and qualification criteria are outlined in Appendix A.

(2) The continuous indication of hydrogen concentration is not required during normal opeartion.

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.

Implementation

Operating license applicants with an operating license date before January 1, 1982 must have design changes completed by January 1, 1882; whereas those applicants with license dates past January 1, 1982 must have all design modifications completed before they can receive their operating license.

Type of Review

A postimplemenation review for applicants for an operating license prior to January 1, 1982 will be performed.

A preimplementation review for applicants for an operating license after January 1, 1982 will be performed.

Documentation Required

Applicants for operating license receiving an operating license before January 1, 1982 will submit documentation before January 1, 1982. Applicants with operating license issued after January 1, 1982, shall provide the required design information at least six months prior to the expected date of operation.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

TVA RESPONSE

(1) Watts Bar has low-volume portable air monitoring equipment with charcoal filters to absorb iodine isotopes. These filters will be analyzed in the health physics laboratory. This capability meets the requirements of II.F.1.

The gaseous effluent monitoring system at Watts Bar Nuclear Plant was designed and constructed to continuously monitor the total gaseous effluent from the reactor, auxiliary and service buildings. Should it become necessary to monitor specific areas of the plant during normal operation or in the event of an accident, the following collection and assessment capabilities are readily available.

Low volume air samplers are onsite and may be located throughout the auxiliary and service buildings in normally occupied areas. These samplers are designed for continuous operation at approximately 1 CFM. In addition, portable low volume air samplers located in the health physics laboratory are available for collecting samples in any specific area of the plant. The above samplers are equipped with filter holders that will accept a two inch (2') charcoal filter cartridge specifically designed for total iodine collection. Additionally, a special Silver Zeolite Radioiodine collection cartridge is available for use during an emergency. Testing of this cartridge indicates that radioactive xenon, krypton and other noble gases are not retained by the Silver Zeolite to interfere with the radioiodine assessment.

To aid in collecting and analyzing radioactive airborne samples in localized areas of the auxiliary building, three portable monitors will be provided and will have the ability to collect and analyze total gaseous effluents. Sample ports are provided in ventilation ducts leading to cubicals that have the potential for high airborne activity, thus allowing samples to be collected from outside the affected area.

To accurately assess the radioiodine collected on the filter, health physics personnel may forward the samples to the radiochemical laboratory. The assessment will be made utilizing a Nuclear Data 6620 computer and three Ge(Li) detectors. Should this system become inoperable, the following alternatives for assessment are available:

a. Gamma spectrometer - Eberline (SAM-2) with two inch diameter NaI detector. This system is used by the health physics unit for emergency environmental monitoring.

b. The training center located on the Sequoyah plant site contains counting equipment identical to the plant radiochemical laboratory. This equipment is available to the plant at all times. Included are one Nuclear Data 6620 computer and two Ge(Li) detectors.

If additional equipment is needed for analysis the samples may be transported to Sequoyah Nuclear Plant. The Sequoyah equipment includes one Nuclear Data 6620 computer and two Ge(Li) detectors.

The plant has procedures for sampling and analysis of in-plant air spaces incorporated in the Health Physics Laboratory Instruction Manual and the Radiation Control Instruction Manual.

Plant health physics technicians are required to complete a formal training program plus receive in-plant training which includes the use of health physics procedures and instrumentation.

(2) The response to Item II.B.3 provides this information.

(3) TVA will comply with these requirements by fuel load except that high level radiation monitors will be located outside the annulus instead of inside containment. Interim measures will be provided in the respective units for quantifying high level releases.

Redundant safety grade high range noble gas effluent monitors will be provided on the shield building vents.

A method or methods of sampling effluent particulates and iodine will be chosen and redundant particulate and iodine effluent sampling systems to the present state-of-the-art will be provided.

The present design has one high range radiation monitor outside the containment in the auxiliary building, opposite the personnel hatch to detect high levels of radioactivity inside the containment. However, its range is not as high as required. Redundant radiation monitors will be provided outside the annulus to meet the NRC's high range requirement. These monitors will be safety grade and will be designed and qualified to function in an accident environment.

<u>Interim Procedures for Quantifying High Level Accidental Radioactivity Releases</u>

To provide interim measures to estimate high level releases, TVA now plans to install a temporary high range detector external to the sampling tubing of the shield building vent monitor. The detector will monitor only gross radioactivity releases and will not be able to distinguish the radioiodine contribution of the total release. TVA will provide a method for easily converting the detector readings and vent flow rate to activity release rates.

Noble Gas Effluent Monitors

A. TVA will provide an instrument to monitor gross releases of radioactivity from the shield building vent. Our present shield building vent monitor provides a gaseous sample for laboratory analysis. Special procedures will be developed for estimating noble gas effluent in the event present instrumentation saturates.

An area radiation monitor with a range of 10^2 mR/hr to 10^7 mR/hr is being placed near the sample piping to the shield building vent monitor assembly. A precalculated relationship between noble gas concentrations in the sample piping, the monitor readings, and the air volume flow rate in the shield building vent will provide an estimate of gross radioactivity release rates. It has been determined that special shielding around the monitor will not be necessary for it to perform its function. This monitor will be functional before exceeding 5% power.

B. By Fuel Load, TVA will provide high range noble gas effluent monitors for all identified release paths. This monitor will meet the requirements of Table II.F.1-3. Information requested on these monitors will be made available to the NRC.

1. Radioiodine and Particulate Effluents

A. A design study to assist in developing interim procedures for monitoring radioiodine and particulate effluents is underway. The procedures will be available for NRC by fuel load.

B. By fuel load, TVA will provide the capability to continuously sample effluents and onsite analysis for radioiodine and particulates with state-of-the-art equipment. The requested information will be made available to the NRC.

1. Containment Radiation Monitors

By fuel load, TVA will provide two radiation monitors outside the annulus which meet the intent of the requirements.

(4) Four qualified, continuous indications of the containment pressure are provided in the main control room. The existing pressure indicators have a range of -1 to 15 psig. Redundant, continuous containment pressure indication with a range up to four times the design pressure (0 to 50 psig) of the steel containment will be provided by fuel load.

The monitors will meet the applicable design requirements for qualification, redundancy and testability in accordance with the Watts Bar design.

(5) The floor of the reactor building serves as the sump for the containment. It is instrumented with four separate,

qualified, and continuous level instruments which indicate in the main control room. The range of the instruments is from less than six inches above the floor up to 20 feet above the floor. If 600,000 gallons of water were introduced into containment in addition to the fluid volume of the reactor coolant system, safety injection accumulators, and a total ice melt, the containment water level would not exceed the 20 ft. range of the level instruments. A small sump suction pocket (about 120 cubic feet) in the reactor building floor serves as a collector for the recirculation piping exiting the containment and does not require qualified level instrumentation.

The narrow range sump level instrument monitors the normal containment sump level and the wide range sump level instrument monitors the emergency sump level.

The wide range sump level instrument meets the applicable requirements for qualification, redundancy, and testability in accordance with the Watts Bar design.

The narrow range sump level instrument meets the appropriate requirements of Regulatory Guide 1.45.

(6) Redundant, safety grade hydrogen analyzers are located in the annulus between the containment and shield building. These monitors provide continuous indication in the main control room within a few minutes of being remotemanually actuated in the main control room. The range of these monitors is from 0 to 10 percent hydrogen concentration from negative 2 psig to positive 50 psig pressure.

Descriptions of the hydrogen analyzer, sampling points read out and system capabilities are provided in FSAR Section 6.2.5 'Combustible Gas Control.'

The hydrogen analyzers meet the applicable requirements for qualification, redundance, and testability in accordance with the Watts Bar design.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

NRC Position

Provide a description of any additional isntrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to privde an unambiguous, easy-to-interpret indication of inadeqnate 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.

Changes to Previous Requirements and Guidance

(1) Specify the 'Design and Qualification Criteria' for the final ICC monitoring system in section, 'Clarification' (items 7, 8 and 9), Attachment 1, and Appendix A.

(2) Specify complete documentation package to allow NRC evaluation of the final ICC monitoring systems.

(3) No preimplementation review is required but postimplementation review of installation and preimplementation review before use as a basis for operator decisions are reqnired.

(4) Installation of additional instrumentation is now required by January 1, 1982.

(5) Clarification item (6) has been expanded to provide applicants with more flexibility and diversity in meeting the requirements for determining liquid level indication by providing possible examples of alternative methods.

Previsous guidance on the disign and qualification criteria for upgrading of existing instrumentation was based on Regulatory Guide 1.97, which is still being developed. Detailed design requirements for incore thermocouples and additional instrumentation were not specified. The pertinent portions of draft Regulatory Guide 1.97 have now been included as Appendix A. Design requirements for incore thermocouples used in the ICC monitoring system are specified in Attachment 1. The only significant change in design requirements involves a relaxation of qualification requirements for display systems amenable to computer processing. This facilitates procurement of computer systems and makes feasible the use of cathode ray tube (CRT) displays that may be needed for porper interpretation of some reactor-water-level systems under development. This relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99% availability for the display systems, by requiring postaccident maintenance accessibility of nonredundant portions of the system, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

The staff has concluded that the previous installation requirement of January 1, 1981 for additional instrumentation is unrealistic for most licensees, due to procurement and development problems associated with proposed measurement methods. Further, the staff cannot find the proposed methods acceptable for use until development programs have been completed.

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., high-void 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 A, '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 A, need not apply to the channel beyond the isolation device if it is designed to provide 99% 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 A, 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) intergration into emergency procedures,

(c) integration into operator training, and

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

Implementation

This requirement must be implemented by January 1, 1982.

Type of Review

A postimplemenation review iwll be performed for installation, and a preimplementation review will be performed prior to use.

Documentation Required

The applicant shall provide a report detailing the planned instrumentation system for monitoring of ICC. The report should contain the necessary information either by inclusion or by reference to previous submittals including pertinent generic reports, to satisfy the requirements which follow:

(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 systems (e.g., subcooling meters and incore thermocouples), including parameter ranges and displays, which provide operating information pertinent to ICC considerations; and

(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 tests porgrams to be conducted for evaluation, qualification, and calibration of additional information.

(4) An evaluation, including proposed actions, on the conformance of the ICC instrument system to this document, including Attachment 1 and Appendix A. 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 isntallation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

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

(8) A summary of key operator action isntructions 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.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.2, 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 capabiity sould be available for <u>all</u> thermocouple temperatures. The range should extend from 200 o#F (or less) to 1800 o# 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 o#F (or less) to 2300 o#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, '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 IE 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 IE, 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 IE.

(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% 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.

INSTRUMENTATION FOR DETECTION OF
INADEQUATE CORE COOLING

TVA RESPONSE

Analysis and procedures for the detection of inadequate core cooling using existing instrumentation have been developed in conjunction with the Westinghouse Owners' Group. This guidance will be incorporated into plant procedures by fuel load.

In addition to the primary method for detecting inadequate core cooling described above, TVA will provide instrumentation to measure water level in the reactor vessel down to the hotleg piping and between the hotleg and bottom of the reactor vessel. Refer to Figure II.F.2-1. This instrumentation will be designed and qualified in accordance with safety grade, Class IE, requirements including redundancy and emergency power.

The Reactor Vessel Level Instrumentation System was designed to provide direct readings of vessel level which can be used by the operator. This Reactor Vessel Level Instrumentation System does not replace existing systems and is not coupled to safety systems, but acts only to provide additional information to the operator.

The Upper Range Reactor Vessel Level Instrumentation has differential pressure measurement across the upper region of the reactor vessel. The system utilizes two differential pressure cells measuring the pressure drop from the reactor coolant hotleg piping to the top of the reactor vessel head. The system provides an indication of reactor vessel water level above the hotleg pipe when the pump in the loop with the hotleg connection is not operating. The number of pumps operating in the other loops has an effect of less than 10 percent of this indication. When the pump is operating in the loop with the hotleg connection, the instrument reading will be offscale.

The narrow range reactor vessel level instrumentation measures vessel level from the top to the bottom of the reactor vessel when only one or no reactor coolant pumps are running. The instrument will also measure the reactor core and internals pressure drop, and therefore an indication of the relative void content or density of the circulating fluid when only one pump is operating. When more than one pump is running, the instrument will be offscale.

The wide range reactor vessel level instrument measures the reactor core internals and outlet pressure drop for any combination of pumps running. Comparison of any measured pressure drop with the measured pressure drop during normal operation will provide an approximate indication of the relative void content or density of the circulating fluid.

To provide the required accuracy for water level measurement, temperature measurements of the reference legs are provided.

These measurements together with the reactor coolant temperature measurements are used to compensate the differential pressure particularly during the environment inside the containment structure following an accident.

The Reactor Vessel Level Instrumentation System utilizes differential pressure cell instrumentation in two of the hotleg pipes. The instrumented hotleg piping will not be adjacent, but with respect to the plant layout, will be on opposite sides of the reactor vessel. The differential pressure cells are to be located outside of containment so that calibration cell replacement, reference leg checks and filling, and operation are made more easily and the overall system accuracy is improved.

Instrumentation for the operator for the Reactor Vessel Level Instrumentation System is intended to be unambiguous and reliable so that operator error or misinterpretation is avoided.

Upper range, narrow range, and wide range level signals are available from each train for display on standard VX-252 type vertical scale voltage meters. Thus, the indication is compatible with existing control board layouts. The indication signals are electrically isolated from the protection set and are suitable to serve as either a standard control grade or postaccident monitoring output.

The control board displays provide the following information:

1. An indication of reactor vessel level (narrow range) for each instrumented set displaying vessel level in percent from 0 to 60 percent after compensation for the effects of the reactor coolant and capillary line temperature and density, when reactor coolant pumps are not operating.

2. An indication of reactor differential pressure (d/p) (wide range) from each instrumented set displaying d/p in percent from 0 to 100 percent, after compensation for the effects of the reactor coolant and capillary line temperature and density effects, when reactor coolant pumps are operating.

3. An indication of upper range vessel level on each of the two instrumented sets displaying vessel level in percent from 60 to 100 percent after compensation for any reactor coolant and capillary line density effects, when the reactor coolant pump in the loop with the hotleg connection is not operating. A status light will indicate the operation of the reactor coolant pump with the hotleg connection.

Redundant displays are provided for the two sets. Level information based on all three d/p measurements is presented. Correction for reference leg densities is automatic. Any error conditions such as out-of-range sensors or hydraulic isolators are automatically displayed on the affected measurements.

The Reactor Vessel Level Instrumentation is to be used in conjunction with a coolant subcooling readout to determine the state and transient behavior of the reactor coolant system. The reactor vessel wide range level indication will read onscale with all four reactor coolant pumps running during normal operation from 0 to 100 percent full power. With all pumps shut down, the indicator will provide a direct indication of water level in the reactor vessell.

Incore Thermocouples

1. The Watts Bar Nuclear Plant incore thermocouples are located at the core exit for each quadrant and, in conjunction with core inlet RTD data, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. Sixteen (four per quadrant) of the core-exit thermocouples will be designated as PAM sensors.

2. The primary operator display is a computer-driven printer. This system has the following capabilities:

   a. A spatially oriented core map is available on demand which indicates the temperature at each core exit thermocouple location.

   b. An example of the Sequoyah selective readings is an on-demand tabular listing of all instantaneous incore thermocouple values.

   c. A printout of average, instantaneous, and maximum values is provided for all T/C temperatures. The range will meet the suggested range of 200°F.

   d. Trend capability showing temperature time histories is designed into the system. Strip chart recorder points are available to assign to any incore thermocouple on demand. In addition, a point value trend printout is available on the control room printer.

   e. Alarm capability if provided in conjunction with the subcooling monitor which uses the average of all the T/C readings in the calculations.

   f. The control room displays are designed for rapid operator access and ease of viewing data. Also, the incore program has a validity-check comparison which reduces the probability of accessing false readings.

3. A backup analog readout is provided with the capability of selective reading of any T/C in the system. The range of the system is 0-700oF.

   Another means of obtaining this data can be obtained by reading the raw signals T/C and reference junction output with portable test equipment. This data is available in the control building and would be accessible under all conditions should the primary and backup display devices fail.

4. This system will be reviewed by the human factors review group as a part of NUREG-0700 task.

5. Conformance to Appendix B.

   The existing system does not meet the requirements of Appendix B. Evaluations are being performed to determine to what extent modifications should be made to upgrade this system. This evaluation, along with an implementation schedule, will be available by fuel loading on Watts Bar unit 1.

6. Present isolation between the primary and backup channels is implemented in the form of electrical switches. The primary and backup display channels are powered by a reliable battery-backed power source.

7. The existing incore T/C system is a very simple set of hardware which should, by virtue of its simplicity, be a highly reliable and accessible system.

8. Same as item 7.

9. Same as item 7.

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

NRC 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:

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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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.

Implementation

This requirement shall be implemented by applicants for operating license prior to the issuance of an operating license.

Documentation Required

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that each of the positions stated above are met. The documentation should include, as a minimum, supporting information including system design description, logic diagrams, electrical schematics, test procedures. and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.1

NUREG-0660, Item II.G.1

NUREG-0694, Part 1

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1980.

EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

TVA RESPONSE

The power-operated relief valves (PORV) and their associated block valves and control components are classified as Class IE and are supplied from the emergency onsite power supply if offiste power is lost. The relief valves and their associated block valves are powered from opposite power trains. All connections to the emergency power supply are through devices that are qualified in accordance with safety grade requirements. For a description of the PORV and block valves, see FSAR Section 5.2.

The pressurizer level indication instrumentation power is taken from the vital power bus (see FSAR Section 7.5). These buses are supplied from the emergency power source when offsite power is unavailable.

Since the Watts Bar design meets NRC recommendations, no changes are anticipated and therefore, the capability to open PORV/block valves will not be affected.

II.K.1.5 REVIEW ESF VALVES

This information is required 4 months before scheduled SER

References

IE Bulletins 79-05
' 79-05A
' 79-06A
' 79-06B
' 79-08

NRC letter of June 26, 1980

REVIEW ESF VALVES

TVA RESPONSE

The information on ESF valve review as provided on the Sequoyah docket (letter dated July 12, 1979, from L. M. Mills to D. B. Vassalo) for Bulletin 79-06A is applicabale to Watts Bar Nuclear Plant.

II.K.1.10 OPERABILITY STATUS

References

IE Bulletins 79-05A
' 79-06A
' 79-06B
' 79-08

NRC letter of June 26, 1980

OPERABILITY STATUS

RESPONSE

NRC concerns with respect to these bulletins are addressed in the responses to NUREG-0737 provided in this report.

II.K.1.17 TRIP PER LOW - LEVEL B/S (See also II.K.2.10)

References

IE Bulletin 79-06A

NRC letter of June 26, 1980.

TRIP PER LOW-LEVEL B/S

RESPONSE

TVA's response to Bulleting 79-06A on Sequoyah Nuclear Plant (letter dated July 12, 1979, from L. M. Mills to D. B. Vassallo) provides applicabale information on trip per low-pressurizer level. The Watts Bar protective logic will cause initiation of safety injection on two out of three low pressurizer pressure signals regardless of pressurizer level. All applicable instructions require manual initiation of safety injection when two of the three pressurizer pressure signals reach the actuation setpoint.

II.K.2.13 THERMAL MECHANICAL REPORT--EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

NRC 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.

Changes to Previous Requirements and Guidance

This requirement has been changed to include all operating pressurized-water reactors (PWRs) and applicants.

NRC 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.

Implementation

Implementation of any modifications will be determined by the results of NRC staff review of the report.

Type of Review

A postimplementation review will be performed.

Documentation Required

Applicants for operating license shall submit the results of their evaluations at least 6 months prior to the issuance of the staff safety evaluation report for a full-power license.

Technical Specification Changes Required

Changes to technical specifications will be determined following staff review.

References

NUREG-0645, Volume 1, Section 2.4.3

Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

Letter from D. G. Eisenhut, NRC, to All Licensees, dated May 7, 1980.

THERMAL MECHNICAL REPORT - EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

TVA RESPONSE

Westinghouse (in support of the Westinghouse Owners Group) is performing a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater for generic Westinghouse plant groups.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

NRC Position

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

Changes to Previous Requirements and Guidance

The previous requirement has been changed to include all PWR operating reactors and applicants.

NRC 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.

The results of this evaluation have been submitted by the B&W licensees and is presently undergoing staff review.

Implementation

Implementation of any modifications will be determined by the results of NRC staff review of the applicants evaluation.

The analysis should be submitted 6 months before the expected issuance date of the staff safety evaluation report for the license.

Type of Review

A postimplementation review will be performed.

Documentation Required

Submit analyses as indicated in 'Implementation.'

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item II.K.2 (C.17)

Letter from R. W. Reid, NRC, to All B&W Operating Plants, dated January 9, 1980.

POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

TVA RESPONSE

The Westinghouse owners' group is addressing the potential for void formation in the rector coolant system (RCS) during natural circulation conditions as described in Westinghouse letter NS-TMA-2298 (T. M. Anderson, Westinghouse, to P. S. Check, NRC dated September 3, 1980). We believe the results of this effort will fully address the NRC requirement for analysis to determine the potential for voiding in the RCS during anticipated transients. A report describing the results of this effort will be provided to the NRC when available.

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

NRC Position

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.

Changes to Previous Requirements and Guidance

The previous requirement has been changed to include all operating pressurized-water reactors (PWRs) and appliants for operating license.

NRC Clarification

This requirement was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from D. F. Ross, Jr., NRC, to all B&W operating plants, dated August 21, 1979.

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

Implementation

Implementation of any modifications will be determined by the results of NRC staff review of this analysis.

The analysis should be submitted 6 months before the expected issuance date of the staff safety evaluation report for a license.

Type of Review

A postimplementation review will be performed.

Documentation Required

Submit analyses as indicated in 'Implementation.'

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0645, Volume 1, Section 2.4.6

Letter from D. F. Ross, Jr., NRC, to All B&W Operating Plants, dated August 21, 1979.

SEQUENTAIL AUXILIARY FEEDWATER FLOW ANALYSIS

TVA RESPONSE

The Westinghouse transient analysis code, LOFTRAN, and the present small break LOCA evaluations analysis code, WFLASH, have both undergone benchmarking against plant information or experimental test facility information. These codes under appropriate conditions have also been compared to each other. The Westinghouse owners' group will provide a report addressing the benchmarking of these codes.

II.K.3.1 INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

NRC 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 assure 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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.

Implementation

If required by action plan item II.K.3.2, licensees shall implement modifications and perform confirmatory tests at the next refueling outage followig staff approval of the design, unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the following refueling outage.

Type of Review

A preimplementation review will be performed.

Documentation Required

If modifications are required, documentation shall include piping and instrumentation diagrams, electrical schematics of design modifications, and an analysis of conformance to IEEE 279-1971 requirements. Documentation shall be submitted by July 1, 1981.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0565, Recommendation 2.1.2.a

NUREG-0611, Recommendations 3.2.4.e and 3.2.4.f

NUREG-0635, Recommendations 3.2.4.a and 3.2.4.b

NUREG-0660

INSTALLATION AND TESTING OF AUTOMATIC
POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

TVA RESPONSE

A report evaluating the operating history of Westinghouse Electric Corporation PORV's was submitted to NRC on March 15, 1981 (WCAP-9804). We have reviewed the report and agree with the Westinghouse determination that the concept of an automatic PORV block valve closure system will provide little additional protection against a PORV LOCA. The post-TMI modifications made to date have significantly reduced the already small probability of a stuck-open PORV LOCA; and it is the position of TVA that, based on the WCAP-9804 study, the automatic PORV isolation system should not be installed at Watts Bar.

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

NRC 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 failiure 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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 in. and 2.0 in. 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 should 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 aplicability 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.

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

The licensee is to submit for staff review a report on the probability of small-break LOCA and safety-valve failure rates along with documentation of applicability (where appropriate).

Technial Specification Changes Required

Changes to technical specifications will not be required.

References

WASH-1400 (NUREG-75/014)

NUREG-0565, Recommendation 2.1.2.d

NUREG-0611, Recommendations 3.2.4.g and 3.2.4.i

NUREG-0635, Recommendation 3.2.4.c

NUREG-0660, Items II.K.3.1, II.K.3.2

REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

TVA RESPONSE

See the response to Item II.K.3.1.

II.K.3.3 REPORTING SV AND RV FAILURES AND CHALLENGES

References

NRC letters of June 26, 1980 and May 7, 1980.

TVA RESPONSE

TVA will promptly report any failure to close of a primary safety or PORV valve or a steam generator safety or atmospheric relief valve. In addition, administrative procedures will be revised to document, in the annual report, all challenges to these valves. In the long term, we recommend this requirement be incorporated in the NRC proposed integrated operational experience reporting system (NUREG/CR-1928).

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

NRC 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 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.

Changes to Previous Requirements and Guidance

Implementation dates are changed to be consistent with test schedule for LOFT test (L3-6) and to provide for blind posttest analysis.

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 requred 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.

Implementation

(1) Document models are to be used for analysis prior to December 3, 1980.

(2) DOE/NRC is to run the LOFT test (L3-6) from December 3, 1980 to December 17, 1980.

(3) NRC will distribute initial conditions approximately 4 weeks after the test.

(4) Prediction results will be submitted approximately 4 weeks after receipt of initial conditions.

(5) NRC determination of model acceptability is due April 1, 1981.

(6) Proposed design modifications (if necessary) are due by July 1, 1981.

(7) Modification (if necessary) is due by March 1, 1982.

Type of Review

An NRC preimplementation review will be performed (if any modifications are required).

Documentation Required

Prediction by vendor analysis of LOFT test (L3-6) is required. Additional information needed will depend upon prediction results.

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.3.2.a

NUREG-0611, Recommendation 3.2.2.a

NUREG-0623

NUREG-0635, Recommendation 3.2.2.a

NUREG-0660

AUTOMATIC TRIP OF REACTOR COOLANT PUMPS
DURING LOSS-OF-COOLANT ACCIDENT

TVA RESPONSE

In response to IE Bulletin 79-06C, Westinghouse (in support of the Westinghouse Owner's Group) performed an analysis of delayed reactor coolant pump (RCP) trip during small break loss-of-coolant accidents (LOCA). This analysis is the basis for the Westinghouse position that an automatic RCP trip is not necessary for a Westinghouse pressurized water reactor since sufficient time is available for manually tripping these pumps. Additionally, the Owner's Group is supporting a Westinghouse best-estimate study using the NOTRUMP computer code to demonstrate that tripping the RCP's at the worst trip time following a small break LOCA will lead to acceptable results and Westinghouse is performing test predictions of the LOFT experiment L3-6.

II.K.3.9 PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION

NRC Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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.

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of an operating license.

Type of Review

A postimplementation review will be performed.

Documentation Required

The applicant shall inform the NRC when the modification has been completed.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0611, Recommendation 3.2.4.b

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

PROPORTIONAL INTEGRAL DERIVATIVE
CONTROLLER MODIFICATION

TVA RESPONSE

The derivation time constant in the pid controller for the pressurizer PORV has been set to OFF (zero) which in effect removes the derivative action from the controller. Removal of the derivative action will decrease the likelihood of opening the PORV since the actuation signal for the valve is then on longer sensitive to the rate of change of pressurizer pressure. This setpoint can be found in the Watts Bar Precautions, Limitations, and Setpoints Document, Revision 0, Page 35, Section 3A.

II.K.3.12 CONFIRM EXISTENCE OF ANTICIPATORY REACTOR TRIP UPON TURBINE TRIP

NRC 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.

Changes to Previous Requirements and Guidance

The date for submittal of design has been extended from July 1, 1980 to January 1, 1981.

NRC Clarification

No further clarification is required.

Implementation

All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evalution report for an operating license.

Type of Review

A preimplementation review will be performed (if design modifications are required).

Documentation Required

All applicants for an operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0611, Recommendation 3.2.4.a

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

CONFIRM EXISTENCE OF ANTICIPATORY
REACTOR TRIP UPON TURBINE TRIP

TVA RESPONSE

Watts Bar has an anticipatory reactor trip on turbine trip feature. FSAR Sections 7.1 and 7.2 will be revised to reflect the 50% permissive on reactor trip following turbine trip. The technical specifications will reflect this revision.

II.K.3.17 REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

NRC 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).

Changes to Previous Requirements and Guidance

This clarification adds the requirement to propose changes that will improve and control availability.

NRC 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. Tests 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.

Implementation

Applicants for operating license should submit their plan for data collection in accordance with the review schedule for licensing.

Type of Review

A postimplementation review will be performed.

Documentation Required

(1) Licensees shall submit a report containing the items noted in the above sections.

(2) Licensees shall submit suggested changes to improve the availability of ECC equipment, if needed.

Technical Specification Changes Required

Changes depend on results of the licensee study.

References

NUREG-0626, Recommendation A.6

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

TVA RESPONSE

TVA will develop and implement a plan for gathering cumulative outage times for ECC equipment before fuel loading. This plan will include:

(1) outage dates and duration of outage
(2) cause of the outage
(3) ECC systems or components involved in the outage, and
(4) corrective action taken.

In the long term, we recommend reporting of ECC outage be incorporated in the NRC proposed integrated operational experience reporting system (NUREG/CR-1928).

REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

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In the long term, we recommend reporting of ECC outage be incorporated in the NRC proposed integrated operational experience reporting system (NUREG/CR-1928).

II.K.3.25 EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON PUMP SEALS

NRC 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.

Changes to Previous Requirements and Guidance

The evaluation and proposed modifications shall be submitted by July 1, 1981. The May 7, 1980, letter called for modifications by January 1, 1982. This clarificaton adds a documentation requirement for the evaluation to be submitted by July 1, 1981. The modification date remains unchanged. Additionally, this task has changed to include Westinghouse and Combustion Engineering operating reactors and operating reactor applicants.

NRC 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.

Type of Review

A preimplementation review of modifications will be performed.

Documentation Required

Applicants for operating licenses shall submit the evaluation and proposals by January 1, 1982, or no later than 6 months prior to expected issuance of the staff safety evaluation report in support of license issuance, whichever is later.

Technical Specification Changes Required

Changes to technical specifications will not be required.

Reference

NUREG-0626, Recommendation B.4

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

EFFECT OF LOSS OF ALTERNATING CURRENT POWER ON PUMP SEALS

TVA 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 exchanges to limit the heat transfer from the reactor coolant to the RCP internals. In the event of 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 either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power for at least two hours.

II.K.3.30 REVISED SMALL-BREAK LOSS-OF-COOLANT-ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

NRC Position

Tha 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.

Changes to Previous Requirements and Guidance

The changed requirement (1) allows for justification of acceptability of present small-break LOCA models by comparison with test data, and (2) requests each licensee to outline scope and schedule for model revision or comparison with test data by late fall, 1980. The original requirement did not allow provision for showing acceptability of present models by comparison with plant data.

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 anlaytical 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 light-water 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, 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.

Type of Review

A postimplementation review of the schedule will be performed. A preimplementation review will be performed by the staff to approve the model and analyses.

Documentation Required

(1) Submit outline of program for model justification.

(2) Licensees shall submit their plant-specific analyses using these models by January 1, 1983, or one year after any models are approved.

(3) Applicants shall submit appropriate information in accordance with the licensing review schedule.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0565, Recommendation 2.2.2a

NUREG-0611, Recommendation 3.2.1a

NUREG-0623

NUREG-0626, Recommendation A.12

NUREG-0635, Recommendation 3.2.1.a and 3.2.5.a

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

TVA RESPONSE

A WFLASH, Appendix K computer code analysis has been provided by Westinghouse. This small break study s applicable to Watts Bar. In addition, the Westinghouse Owner's Group has responded to this item. If further analyses are required using the more advanced Westinghouse NOTRUMP small break model these will be provided at a later date.

II.K.3.31 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR PART 50.46

NRC 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC Clarification

See 'Clarification' for item II.K.3.30

Implementation

Calculations shall be submitted by January 1, 1983, or 1 year after staff approval of LOCA analysis models, whichever is later, only if model changes have been made.

Type of Review

A review for conformance with 10 CFR 50.46 limits will be performed.

Documentation Required

Operating License Applicants--All applicants for operating license should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

Technical Specification Changes Required

Changes to technical specifications are to be determined.

References

NUREG-0565, Recommendation 2.2.2.b

NUREG-0611, Recommendation 3.2.1.b

NUREG-0626, Recommendations A.13 and B.10

NUREG-0635, Recommendation 3.2.1.b

Letter from D. G. Eisenhut, NRC, to All Operating Licensees, dated May 7, 1980.

PLANT SPECIFIC CALCULATIONS TO SHOW
COMPLIANCE WITH 10 CFR PART 50.46

TVA RESPONSE

See the response to Item II.K.3.30.

III.A.1.1 IMPROVING LICENSEE EMERGENCY PREPAREDNESS--LONG-TERM

NRC Position

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 Preparation in Support of Nuclear Power Plants.'

Changes to Previous Requirements and Guidance

The final regulations on emergency planning (45 FR 55401-55413) which become effective on November 3, 1980, require the submittal and implementation of the radiological emergency response plans of licensees and state and local entities within the plume exposure and ingestion emergency planning zones (EPZ).

NUREG-0654 has been revised to include changes developed from team reviews and comments obtained during the comment period.

The revised NUREG-0654 establishes the schedule for installation of meteorological equipment to meet a prescribed implementation date (also see proposed Revision 1 to Regulatory Guide 1.23). The NRC rule establishes July 1, 1981, as the date when the prompt notification capability is to be functional. Item III.A.1.2 establishes dates when emergency response facilities must be functional.

NRC Clarification

In accordance with Task Action Plan item III.A.1.1, '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 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 delineated 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).

Implementation of the new rules levied the requirement for the licensee to provide procedures implementing the upgraded emergency plans to the NRC for review. Revision 1 to NUREG-0654 (FEMA-REP-1) 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 emergency response facilities and establishes firm dates 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, 'Criteria 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 50.47). 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 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 foodstuff requiring protective measures and may also provide information to determine the necessity for sheltering and 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 accessible 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 effluent 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-06654 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.

Implementation

For operating license applicants the following implementation milestones shall be met to address the four basic elements of the introduction to Appendix 2 to NUREG-0654.

Milestones are numbered and tagged with the following code; a-date, b-activity, c-minimum acceptance criteria. They are as follows:

(1) a. Four months before scheduled safety evaluation report

b. Submital of radiological emergency response plans

c. A description of the plan to include elements of NUREG-0654, Revision 1, Appendix 2

(2) a. Four months before SER

b. Submittal of implementing procedures

c. Methods, systems, and equipment to assess and monitor actual or potential offsite consequences of a radiological emergency condition shall be provided

(3) a. Upon receipt of operating license

b. Implementation of radiological emergency response plans

c. Four elements of Appendix 2 to NUREG-0654 with the exception of the Class B model of element 3, or

Alternative to item (3) requiring compensating actions:

A meteorological measurements program which is consistent with the existing technical specifications as the baseline or an element 1 program and/or element 2 system of Appendix 2 to NUREG-0654, or two independent element 2 systems shall provide the basic meteorological parameters (wind direction and speed and an indicator of atmospheric stability) on display in the control room. An operable dose calculation methodology (DCM) shall be in use in the control room and at appropriate emergency response facilities. The following compensating actions shall be taken by the license for this alternative:

(i) if only element 1 or element 2 is in use:

- o# The licensee (the person who will be responsible for making offsite dose projections) shall check communications with the cognizant National Weather Service (NWS) first order station and NWS forecasting station on a monthly basis to ensure that routine meteorological observations and forecasts can be accessed.

- o# The licensee shall calibrate the meteorological measurements program at a frequency no less than quarterly and identify a readily available source of meteorological data (characteristic of site conditions) to which they can gain access during calibration periods.

- o# During conditions of measurements system unavailability, an alternate source of meteorological data which is characteristic of site conditions shall be identified to which the licensee can gain access.

- o# The licensee shall maintain a site inspection schedule for evaluation of the meteorological measurements program at a frequency no less than weekly.

- o# It shall be a reportable occurrence if the meteorological data unavailability exceeds the goals outlined in Proposed Revision 1 to Regulatory Guide 1.23 on a quarterly basis.

(ii) The portion of the DCM relating to the transport and diffusion of gaseous effluents shall be consistent with the characteristics of the Class A model outlined in element 3 of Appendix 2 to NUREG-0654.

(iii) Direct telephone access to the individual responsible for making offsite dose projections (Appendix E to 10 CFR Part 50(IV)(A)(4)) shall be available to the NRC in the event of a radiological emergency. Procedures for establishing contact and identification of contact individuals shall be provided as part of the implementing procedures.

This alternative shall not be exercised after July 1, 1982. Further, by July 1, 1981, a functional description of the upgraded programs (four elements) and schedule for installation and full operational capability shall be provided (see milestones 4 and 5).

(4) a. March 1, 1982

b. Installation of Emergency Response Facility hardware and software

c. Four elements of Appendix 2 to NUREG-0654, with exception of the Class B model of element 3.

(5) a. July 1, 1982

b. Full operational capability of milestone 4.

c. The Class A model (designed to be used out to the plume exposure EPZ) may be used in lieu of a Class B model out to the ingestion EPZ. Compensating actions to be taken for extending the application of the Class A model out to the ingestion EPZ include access to supplemental information (meso and synoptic scale) to apply judgment regarding intermediate and long-range transport estimates. The distribution of meteorological information by the licensee should be as follows by July 1, 1982:

Meteoroglical Information	CR	TSC	EOF	NRC and Emergency Response Organizations
Basic Met. Data (e.g., 1.97 Parameters)	X	X	X	X (NRC)
Full Met. Data (1.23 Parameters)		X	X	X
DCM (for Dose Projections)	X	X	X	X
Class A Model (to Plume Exposure EPZ)	X	X	X	X
Class B Model or Class B Model (to Ingestion EPZ)		X	X	X

(6) a. July 1, 1982, or at the time of the completion of milestone 5, whichever is sooner.

b. Mandatory review of the DCM by the licensee

c. Any DCM in use should be reviewed to ensure consistency with the operational Class A model. Thus, actions recommended during the initial phases of a radiological emergency would be consistent with those after the TSC of EOF are activated.

(7) a. September 1, 1982

b. Description of the Class B model provided to the NRC

c. Documentation of the technical bases and justification for selection of the type Class B model by the licensee with a discussion of the site-specific attributes.

(8) a. June 1, 1982

b. Full operational capability of the Class B model

c. Class B model of element 3 of Appendix 2 to NUREG-0654, Revision 1

Schedule for Near-Term Operating Licenses--For applicants for an operat ing license, at least milestones 1, 2, and 3 shall be met prior to the issuance of an operating license. Subsequent milestones shall be met by the same dates indicated for operating reactors. For the alternative to milestone 3, the meteorological measurements program shall be consistent with the NUREG-75/087, 'Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,' Section 2.3.3 program as the baseline or element 1 and/or element 2 systems.

Type of Review

A postimplementation review will be performed.

Documentation Required

Complete updated emergency plans and complete implementing procedures shall be submitted four months prior to scheduled SER.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-75/087

NUREG-0654 (FEMA-REP-1), Revision 1

NUREG-0696

Regulatory Guide 1.23, Proposed Revision 1

IMPORVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM

TVA RESPONSE

TVA has met milestones 1 through 4 and will address the remaining items in accordance with the revised requirement date of January 1, 1982.

III.A.1.2 UPGRADE EMERGENCY FACILITIES

The requirements concerning Technical Support Center (TSC). Operatinal Support Center (OSC) and Emergency Operations Facility (EOF) are pending final version of NUREG-0696.

UPGRADE EMERGENCY SUPPORT FACILITIES

TVA RESPONSE

The details of the proposed facilities are being evaluated for Watts Bar. The following are the proposed implementation dates:

1. TVA's centralized emergency control center is fully operational.

2. The local recovery center (LRC) shall be implemented and Watts Bar Radiological Emergency Plan (REP) revised to reflect this facility by fuel load.

3. Access to meteorological and radiological data, using the Chattanooga Central Emergency Control Center (CECC) central processor, shall be implemented at all appropriate emergency centers by fuel load.

4. The Watts Bar technical support center (TSC) will be operational by fuel load. Upgrading of the TSC is being evaluated.

III.D.1.1 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS

NRC 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 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.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside 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 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.

Implementation

This requirement shall be implemented by applicants for operating license prior to issuance of a full-power license.

Documentation Required

Applicants shall submit the information requested in the 'Clarification' section of this position at least four months prior to issuance of a fuel-loading license.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.6.a

NUREG-0660, Item III.D.1.1

NUREG-0694, Part 2

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated October 17, 1979.

INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED WATER REACTORS

TVA RESPONSE

Plant design was reviewed to evaluate ways to minimize radioactive fluid leakage. Plant systems that were reviewed included RHR, containment spray, safety injection (recirculation mode), CVC, sampling, and waste disposal. The examination included valve stem packing leakoffs, rotating seals, gasket connections, vents, and drains.

As a result of the review, a second pressure boundary will be incorporated on about twenty vents and drains found on pump suction lines and pump casings. The second pressure boundary will be a second valve in most cases and an occasional blind flange.

An additional review was conducted with regard to the North Anna 1 incident and no similar release path was found. The Watts Bar design routes the overpressure relief from the volume control tank to the pressurizer relief tank and all relief paths from high pressure systems vent back into containment to the pressurizer relief tank. All tanks containing radioactivity in the radwaste system and the CVCS vent to a contained release path which is continuously monitored.

TVA will identify the above systems that may be leak checked and will implement a periodic leak check program on these systems. System leakages will be reported to the NRC.

Procedures for reducing and quantifying leakage from liquid systems will be provided. These procedures were written in compliance with the guidelines listed below.

1. Visual inspection with the system in operation is required.

2. Closed loop systems, such as component cooling water, will not be inspected.

3. Inspection will be performed annually.

4. Leakage will be quantified and specifically located by valve number, pump flange or other similar means.

5. Leakages will require immediate attention. All leakage identified will be 'tracked' in plant until the leakage is stopped or controlled (i.e., normal pump seal leakage per manufacturer's spec).

6. Initial leak test results will be provided to the NRC.

The systems identified for leakage checks are listed below.

a. Safety Injection
b. Containment Spray
c. RHR
d. Chemical and Volume Control
e. Sampling

Identification of gaseous leakage is accomplished in response to any alarm from area radiation detectors. Leakages will require immediate attention. All gaseous leakages will be 'tracked' and be controlled. The leak testing procedures for the waste gas system are in SI 656.

The results of the leak reduction program for liquid and gas systems are in Item III.D.1.1, Attachment A.

III.D.3.3 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

NRC Position

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

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

NRC 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:

(1) 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.

(2) Unanticipated isolated 'hot spots' may occur in locations where no stationary monitoring instrumentation is located.

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

(4) 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.

Applicants must provide by fuel loading the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer (SCA). The SCA window should be calibrated to the 365 KeV of iodine-131 using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

Implementation

Applicants for fuel-loading license shall meet this position prior to fuel loading.

Type of Review

A postimplementaiton review will be performed.

Documentation Required

For applicants for an operating license, provide a description of the in-plant airborne radioiodine sampling and analysis systems specifying the number and types of samplers, sample media, sample flushing methods, and sample analysis equipment type and location.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0578, Recommendation 2.1.8.c

NUREG-0660, Item III.D.3.3

Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, dated September 13, 1979.

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

IMPROVED INPLANT IODINE INSTRUMENTAITON UNDER ACCIDENT CONDITIONS

TVA RESPONSE

See the responses to Items II.B.3 and II.F.1.

III.D.3.4 CONTROL-ROOM HABITABILITY REQUIREMENTS

<u>NRC Position</u>

In accordance with Task Action Plan item III.D.3.4 and control room habitability , licensees shall assure 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).

<u>Changes to Previous Requirements and Guidance</u>

There are no changes to the previous requirements.

<u>NRC Clarification</u>

(1) Applicants 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 requiremetns are met. Applicants may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

(2) All applicants 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;
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) Regualtory Guide 1.78, 'Assumptions for Evaluating the Habitability of Regulatory 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 Accident 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.

Applicants shall submit the results of their findings as well as the basis for those findings. In providing the basis for the habitability finding, applicants may reference their past submittals. Applicants should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1 is provided.

(3) All applicants with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references.

These applicants shall perform the necessary evaluations and identify appropriate modifications.

Each applicant's 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 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 applicant 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.

Implementation

Applicants for operating licenses shall submit their responses prior to issuance of a full-power license. Modifications needed for compliance with the control-room habitability requirements specified above should be identified, and a schedule for completion of the modifications should be provided. Implementation of such modifications should be started without awaiting the results of the staff review. Additional needed modifications, if any, identified by the staff during its review will be specified to applicants.

Type of Review

A postimplementation review will be performed.

Documentation Required

Applicants for an operating license shall submit their responses prior to full-power licensing.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660, Item III.D.3.4

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

III.D.3.4 ATTACHMENT 1, INFORMATION REQUIRED FOR CONTROL-ROOM HABITABILITY EVALUATION

(1) Control-room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release

(2) Control-room characteristics

(a) air volume control room

(b) control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)

(c) control-room ventilation system schematic with normal and emergency air-flow rates

(d) infiltration leakage rate

(e) high efficiency particulate air (HEPA) filter and charcoal adsorber efficiencies

(f) closest distance between containment and air intake

(g) layout of control room, air intakes, containment buidling, and chlorine, or other chemical storage facility with dimensions

(h) control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.

(i) automatic isolation capability-damper closing time, damper leakage and area

(j) chlorine detectors or toxic gas (local or remote)

(k) self-contained breathing apparatus availability (number)

(l) bottled air supply (hours supply)

(m) emergency food and potable water supply (how many days and how many people)

(n) control-room personnel capacity (normal and emergency)

(o) potassium iodide drug supply

(3) Onsite storage of chlorine and other hazardous chemicals

(a) total amount and size of container

(b) closest distance from control-room air intake

(4) Offsite manufacturing, storage, or transportation facilities

of hazardous chemicals

(a) identify facilities within a 5-mile radius;

(b) distance from control room

(c) quantity of hazardous chemicals in one container

(d) frequency of hazardous chemical transportation traffic (truck, rail, and barge)

(5) Technial specifications (refer to standard technical specifications)

(a) chlorine detection system

(b) control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

CONTROL ROOM HABITABILITY REQUIREMENTS

TVA RESPONSE

The Watts Bar Control Room evaluation based on Regulatory Guide 1.78, Regulatory Guide 1.95, and General Design Criteria 19 is specifically addressed in FSAR Section 6.4.

The information required by this item is included in FSAR Section 2.2, 6.4, 6.5, 12.3, 15.5, and technical specifications. Also, the design and operation of the Watts Bar Control Room is essentially the same as the control room at Sequoyah Nuclear Plant with any local differences being minor changes to alleviate individual problems at either plant.