TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 29, 1981

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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

My letter to you dated September 14, 1981 provided TVA responses to NRC concerns as specified in NUREG-0737 for Watts Bar Nuclear Plant. Enclosed are revised responses to the following items.

I.A.1.2	II.B.1	II.F.2	II.K.3.30
I.C.2	II.B.3	II.K.2.13	II.K.3.31
I.C.4	II.D.1	II.K.2.17	III.D.3.4
I.C.5	II.E.4.2	II.K.2.19	
I.C.6	II.F.1	II.K.3.5	

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 29th day of October 1981
Paulette H. White
Notary Public
My Commission Expires 9-5-84

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Enclosure

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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 engineer's normal work station is in the control room, but he periodically makes inspections of plant equipment. He 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.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

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 before fuel loading to evaluate the effectiveness of the turnover procedures.

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

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, before fuel loading, 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, before fuel loading, 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 in 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. Component Cooling Water
G. Containment Spray
H. Residual Heat Removal
I. Emergency Diesel Generators
J. Upper Head Injection

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.

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*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.

TVA RESPONSE

Design Basis

The basic function of the Reactor Vessel Head Vent System (RVHVS) is to remove noncondensible 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 noncondensible gases in the Reactor Coolant System (RCS).

General Description

The RVHVS is designed to remove noncondensible gases or steam from the RCS via remote manual operations from the control room. The RVHVS is connected above the reactor vessel to one of the Upper Head Injection System discharge pipes. The system discharges either into the pressurizer relief tank or directly into a well ventilated containment area. 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 flow diagram of the RVHVS is shown in Figure 1. The system consists of two 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 equipment design parameters are listed in Table 1.

The active portion of the RVHVS consists of four one-inch solenoid operated valves. The inboard solenoid operated valves are open/close isolation valves. The outboard valves, located in the branch lines to the pressurizer relief tank and to the containment, are remotely operated throttle valves capable of regulating the flow rate through the system. With two valves in series in each flow path, the possibility of reactor coolant pressure boundary leakage is minimized. 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. Similarly, the throttle valve in one branch is powered by one vital power supply and the throttle valve in the second branch line is powered by a second vital power supply. They are also fail closed, normally closed valves. The isolation and throttle valves are included in the Westinghouse valve operability program which is an acceptable alternative to Regulatory Guide 1.48. These valves will be qualified to IEEE Standard 323-1974, 344-1975, and 382-1972.

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 nor 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 a spuriously opened flow path due to the spurious movement of one valve. As such, power lockout to any valve is not considered necessary.

The RVHVS valves are operated from the control room. The isolation valves have stem position switches. The position indicator for each isolation valve is monitored in the control room by status lights. The throttle valve position is monitored by a linear voltage differential transmitter.

The reactor vent piping is orificed to 3/8 of an inch. This orifice forms the Safety Class 1 to the 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 of the centrifugal charging pumps.

A break of the RVHVS line upstream of the orifice would result in an intermediate LOCA from the UHI pipes. Such a break is similar to those analyzed in the FSAR.

The system provides for venting the reactor vessel head by using only safety grade equipment. From the orifice to the first anchor downstream of the throttle valves, all equipment is designed and fabricated in accordance with ASME Code, Section III, Class 2 requirements. The remainder of the piping is non-nuclear safety. The RVHVS satisfies applicable requirements and industry standards including the ASME Code classification, safety classification, single-failure criteria, and environmental qualification.

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.

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.

TABLE 1

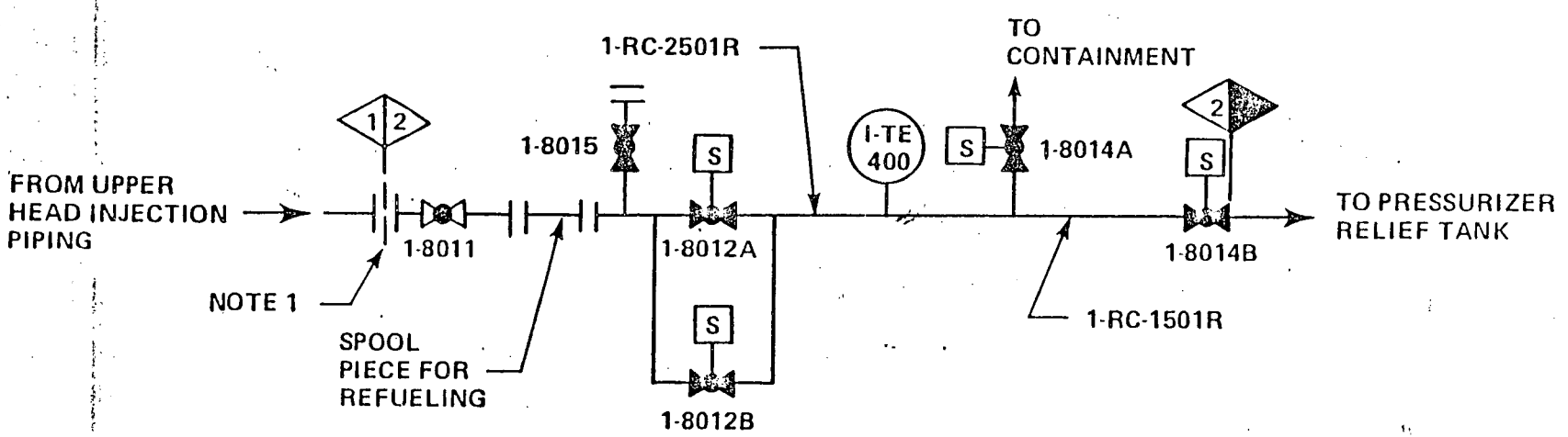
REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT
DESIGN PARAMETERS

Valves

Number (includes one manual valve)	5
Design pressure, psig	2485
Design temperature, ^{o}F	650

Piping

Vent line, nominal diameter, in.	1
Design pressure, psig	2485
Design temperature, ^{o}F	650
Maximum operating temperature, ^{o}F	620



NOTE 1 - FLOW RESTRICTION ALLOWS TRANSITION FROM ANSI SAFETY CLASS 1 TO 2

Figure 1. Reactor Vessel Head Vent System

20382-1

II.B.3 POSTACCIDENT SAMPLING CAPABILITY

NRC Position

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

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high-fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

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

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) Allows combined time of 3 hours or less for sampling and analysis.

(2) Specifies that licensee may use online sampling and analysis to meet the 3-hour time requirement but must provide capability to remove grab samples of reactor coolant and containment atmosphere for separate analysis.

(3) Implementation date has been changed to January 1, 1982.

(4) Provides design guidance for sampling and analytical capability.

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.

Implementation

Installation should take place by January 1, 1982.

Type of Review

A postimplementation review will be performed.

Documentation Required

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.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

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

Item 1. In compliance with the requirements of NUREG-0737, II.B.3, TVA is implementing a post-accident sampling facility (PASF) capable of sampling and analyses of reactor coolant and containment air as well as providing for grab samples to be taken for offsite analyses. The equipment has been furnished by Sentry Equipment Company. TVA is also providing a radchem laboratory for radiological and chemical analyses.

Item 2. Sufficient shielding shall be provided to meet the requirements of GDC-19 (assuming NUREG 1.4 source terms). A design criteria document is being drafted by TVA to ensure this.

Item 3. It is the intent of TVA to meet the requirements of sampling and analysis outlined in Regulatory Guide 1.97, Revision 2.

Item 4. Our design documents stipulate that all equipment in the PASF which may be required to be operational during post-accident operations shall have access to two independent offsite power supplies. Consequently, all electrically powered components associated with post-accident sampling are capable of being operationally restored within 30 minutes of an accident involving core damage in the event of loss of one source of offsite power.

Item 5. Containment isolation valves shall be environmentally qualified in accordance with the requirements in IEEE Standard 323-1974, which includes qualification documentation, as stated in NUREG-0588.

Item 6. Our procedure for relating radionuclides in reactor coolant to extent and kind of core damage is in accordance with the requirements in Regulatory Guide 1.97. TVA has the capability to quantify in less than three hours (1) noble gas radionuclides which suggest fuel cladding failure, (2) iodine and cesium isotopes which correlate to high fuel centerline temperatures, and (3) nonvolatile isotopes which indicate fuel melting. We utilize gamma spectroscopy with other standard radiochemical methods to accomplish this. Procedures providing direction for sample collection, transfer, and analysis will be completed before fuel loading.

Item 7. High-pressure carrier gas, associated with the gas chromatograph, cannot enter the reactor coolant system. The online gas analysis equipment has a parallel interface with the containment atmosphere sample panel. (Note: There is a clear distinction between online equipment and inline equipment. The latter is in constant contact with the process fluid to be sampled and analyzed. Online equipment can remotely sample, transfer, and subsequently analyze the fluid without continued interfacing with the process fluid).

Item 8. Online analysis of dissolved oxygen (DO) and chlorides in the reactor coolant can be performed in the 0.1 to 20 ppm range for both species. However, for DO levels less than 0.1 ppm, a laboratory analysis is required.

Item 9a. Our design documents state the design life of all major components, equipment, and instrumentation shall be 40 years and be able to survive an accident dose in accordance with Regulatory Guide 1.97. Items designed for post-accident service shall be designed to remain functional in post-accident environments for at least three years. Insofar as testing frequency, only routine standardization of test equipment according to manufacturers' recommendations are necessary.

Item 9b. Our design documents indicate the capability to perform a complete set of post-accident sampling and analyses, to verify equipment readiness, and to train personnel without affecting normal power production maneuvers. (Operator training for PAS is being provided by NUS, who has been subcontracted by the PAS vendor, Sentry Equipment Company).

NOTE: NUC PR and EN DES should review this response for compliance on Watts Bar.

The implementation date specified by the NRC is January 1, 1982; EN DES should provide a specific date for completion of modifications. If this is after January 1, 1982, EN DES should provide justification for the new date.

NUC PR should provide confirmation of Technical Instructions for Watts Bar, which contain procedures similar to TI-16 and TI-66 for Sequoyah. NUC PR should revise this response as appropriate to include correct references.

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 for BWRs and July 1, 1982 for PWRs:

(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--April 1, 1982
Plant-specific submittals confirming adequacy of safety and relief valves based on applicant preliminary review of generic test program results--April 1, 1982
Plant-specific reports for safety and relief valve qualification--July 1, 1982
Plant-specific submittals for piping and support evaluations--July 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

Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Generic Letter 81-36 from D. G. Eisenhut dated September 29, 1981.

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.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.

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

TVA RESPONSE

TVA REPONSE TO NUREG-0737

WATTS BAR NUCLEAR PLANT

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 coincident 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.

   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 postaccident mitigation are operational. Table II.E.4.2-1 shows the different isolation signals and the parameters that initiate each signal.

   Isolation is provided on the following three levels:

1. Nonessential systems - These systems are not required for postaccident mitigation. They are isolated automatically upon receipt of a Phase A isolation signal and, except for some instrument lines, have two isolation barriers as required by the GDC. Manual isolation valves are sealed closed as required by S.R.P. 6.2.4.

2. Essential systems - This group consists of the emergency core cooling systems, the containment spray system, and postaccident 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 (Table 1). 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 postaccident and (2) that each containment isolation valve received the proper isolation signal.

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.

The containment isolation system is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. Prior to reset of the initiating signal, a valve closed by the signal can be opened only by constant operator demand with a valve's individual handswitch. The valve will return to the containment isolation position when the operator releases the handswitch. 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.

5. The Watts Bar containment pressure high setpoint is required to be less than or equal to 1.54 lb/in^2g by Technical Specification 3.3.2.1, 'Engineering Safety Feature Actuation system Instrumentation.' It is more than 1 lb/in^2g above the highest primary containment internal pressure allowed by Technical Specification 3.6.1.4, 'Containment Internal Pressure'; however, TVA believes that the present containment pressure high setpoint is adequate. The reasons are stated

below:

The containment pressure high channel actuates safety injection, turbine trip, and feedwater isolation. Phase 'A' containment isolation is actuated from the safety injection logic. Reducing the containment pressure high setpoint increases the potential not only for inadvertent containment isolation but also inadvertent safety injection and feedwater isolation.

Adequate protection is already provided to prevent the release of radioactive materials following an accident. Phase 'A' isolation, which includes containment ventilation isolation, is initiated by diverse signals, including all safety injection actuation channels, including low pressurizer pressure, high gaseous or particulate activity in containment. Further containment ventilation isolation is initiated by high activity in the purge air exhaust.

The containment pressure high and high-high setpoints and low pressurizer pressure setpoints are reached almost immediatley in large loss of coolant accidents (LOCA). Lowering the containment pressure high setpoint will not provide any additional safety margin to the accident analyses because of the speed at which the containment pressure rises for large LOCA's. Two cases for small LOCA's are considered: those for which the charging system can maintain Reactor Coolant System inventory and those that cannot. In the first case, core uncovery will not occur and the radioactivity released is limited to the material contained in the coolant. Containment vent isolation will occur. The setpoint for the ventilation isolation signals are set to prevent releases exceeding 10 CFR Part 20 limits (normal release limits). Lowering the containment pressure high setpoint will not provide any significant additional safety margin.

In the second case, safety injection will always occur much sooner than core uncovery. Since phase 'A' isolation occurs on safety injection, radioactivity releases prior to isolation are limited to material contained in the coolant. Containment vent isolation will occur. Lowering the containment pressure high setpoint will not provide any significant additional safety margins.

TVA believes that the present containment pressure high setpoint of 1.54 lb/in^2g is adequate. Reduction of this setpoint would provide no significant additional safety margin. Instead, it would increase the potential for inadvertent containment isolation and safety injection.

6. TVA will address this issue in our response to staff question 22.49.

7. 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 (Table II.E.4.2-1). At present, the isolation of nonessential 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.

TABLE II.E.4.2-1

Isolation Signals

Phase A

Initiates - Manually - 1 of 2 handswitches or,
Manually - SIS switch or,
Automatically - SIS auto initiation

SIS Initiation

Manuallly - 1 of 2 handswitches or,
Automatically - on 2 out of 3 high containment pressure or,
- 2 out of 3 logic on any of 4 sets of differential pressure between steam lines or,
- low pressurizer pressure on any of 3 channels
- coincident high steam line flow with low steam line pressure or low-low average RCS temperature. Each loop has two high flow meters. One pressure and temperature instrument is provided per loop. At least 2 of the 4 loops must reach the instrument setpoints to initiate the SIS.

Phase B Initiation

Manually - 2 of 4 handswitches or,
Automatically - 2 of 4 high-high containment pressure

Containment Ventilation Initiation

Manually - Phase A manual initiate or,
- Phase B manual initiate or,
- SIS manual initiate or,
Automatically - SIS auto-initiate or,
- high radiation
1 sensor (train A only) or,
- high radiation
1 sensor (train B only) or,
- high purge exhaust radiation
1 of 2 sensors

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.

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# Ci/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# Ci/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# Ci/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.

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.

NRC Position

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

Changes to Previous Requirements and Guidance

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

(1) Specifies a lower range so that the monitor can follow the radiation increase from low levels of radiation for personnel safety up to the maximum expected in major accidents;

(2) Specifies that monitors be located in containment to view a large segment of the containment atmosphere which will more accurately reflect and monitor accident conditions;

(3) Specifies accuracy and energy response in order to ensure accurate measurements independent of the energy spectrum of an accident (this specification was referenced in the letter of October 30, 1979 in referencing Regulatory Guide 1.97, Rev. 2);

(4) Specifies design and qualification criteria to ensure that the monitor will function in an accident environment;

(5) Specifies that electronic calibration is acceptable for higher dose rate ranges because such methods are sufficient to provide acceptable accuracy;

(6) Deletes the requirement for NRR (Office of Nuclear Reactor Regulation) preimplementation review if the monitors meet the listed specifications because the monitor specifications ensure that adequate monitors will be installed;

(7) Moves the implementation date to January 1, 1982 because of the potential unavailability of appropriate equipment and because the qualification of monitors is incomplete;

(8) Requires documentation of alternative proposals for monitors that do not meet the requirements of Table II.F.1-3.

NRC Clarification

(1) Provide two radiation monitor systems in containment which are documented to meet the requiremetns of Table II.F.1-3.

(2) The specification of 10 8#rad/hr in the above position was based on a calculation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA

(loss-of-coolant accident) containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10 7#R/hr.

(3) The monitors shall be located in containment(s) in a manner 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 description of the installed high-range containment monitors and specify the locations of these monitors inside containment. The description 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.

References

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.

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 Reguirements 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.

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.

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.

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.

By initial fuel loading for each unit, TVA will provide redundant high range radiation monitors in the upper and lower compartments inside containment at Watts Bar. The design of these monitors is essentially identical to the monitor design for Sequoyah as described in L. M. Mills' letter to A. Schwenser dated July 8, 1980 and L. M. Mills' letter to E. Adensam dated July 2, 1981.

Interim Procedures for Quantifying High Level Accidental Radioactivity Releases

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 inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

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 required.

(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.

(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.

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 from the top to the 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 by Westinghouse 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 0-700°F.

   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.