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

August 12, 1982

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 letters to you dated September 14, 1981 (original submittal) and October 29, 1981 and March 11, and June 11, 1982 (revised responses to certain items) provided TVA's responses to NRC concerns specified in NUREG-0737 for Watts Bar Nuclear Plant. Enclosed are revised responses to items II.B.1, II.B.2, II.F.1, and II.F.2.

If you have any questions concerning this matter, please get in touch with D. A. Kulisek at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Licensing

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Sworn to and subscribed before me
this 12th day of August 1982.
Paulette H. White
Notary Public

My Commission Expires 9-5-84

Change:	L	E
	1	NP
PDR	1	NP
LPDR	1	NP
NTIS	1	NP
NSIC	1	NP

Enclosure
cc: U.S. Nuclear Regulatory Commission
Region II
Attn: Mr. James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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50-390 WATTS BAR 1 &

REVISED RESPONSES TO NUREG-0737 ITEMS II.B.1,II.B.2,II.F.1 & WESTINGHOUSE APPLICATION FOR WITHOLDING PROP SUMMARY REPORT "WESTINGHOUSE REACTOR VESSEL INSTRUMENTATION SYS FOR MONITORING LTR.DTD.8/12/82 8208200198

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Date 8/12/82

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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.

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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.

(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.

NRC Clarification

A. General

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

(2) Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

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

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

(5) A positive indication of valve position should be provided

in the control room.

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

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

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

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

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

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

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

C. PWR Vent Design Considerations

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

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

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

Implementation

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

Type of Review

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

Documentation Required

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

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

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

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

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

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

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

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

REACTOR COOLANT SYSTEM VENTS

TVA RESPONSE (Revised August 12, 1982)

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.

With regard to venting of the pressurizer, TVA has decided to replace the presently installed relief valves with qualified solenoid-operated valves similar to those in the RVHVS to provide a qualified venting path. This modification will be implemented in lieu of a separate pressurizer vent system.

TVA will install the RVHVS and new pressurizer relief valves 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

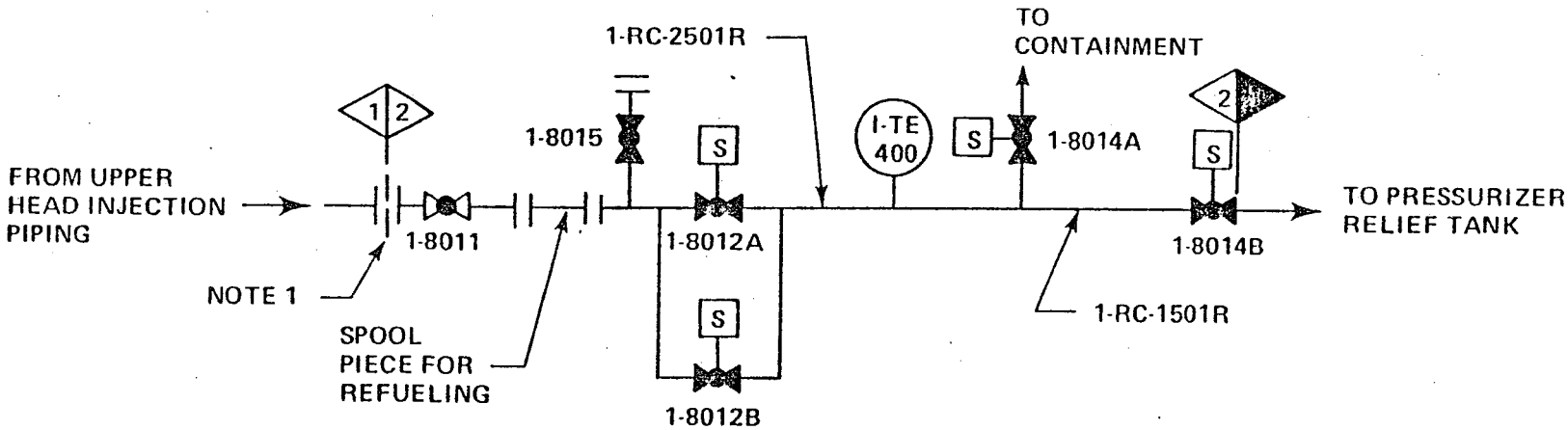
REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT
DESIGN PARAMETERS

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

NRC Position

With the assumption of a postaccident release or radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

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

Changes to Previous Requirements and Guidance

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

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

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

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

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

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

above order.

(6) Because of difficulty in obtaining equipment (e.g., remote-operated valves), the implementation date is moved to January 1, 1982, or the first outage of sufficient duration thereafter, but no later than July 1, 1982.

NRC Clarification

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

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

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

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

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

(1) Source Term

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

occurs before fission products can be transported to various systems).

(a) Liquid-Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of all others are assumed to be mixed in the reactor coolant and liquids recirculated by Residual Heat Removal (RHR), High-Pressure Coolant Injection (HPCI), and Low-Pressure Coolant Injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, you may assume that the water contains no noble gases.

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

(2) Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation should include, but not be limited to, containment, Residual Heat Removal System, Safety Injection Systems, Chemical and Volume Control System (CVCS), Containment Spray Recirculation System, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under Item III.D.1.1, 'Integrity of Systems Outside Containment Likely To Contain Radioactive Material for PWRs and BWRs.' Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Item III.D.1.4, 'Radwaste System Design Features To Aid in Accident Recovery and Decontamination.'

(3) Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore,

allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case bases. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident.

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

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

(4) Radiation Qualification of Safety-Related Equipment

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

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

(b) LOCA events in which the primary system may not depressurize should consider the source term (100% noble gases, 50% iodines, and 1% particulate) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10% noble gases, 10% iodines, and 0% particulate as a source term.

The following table summarizes these considerations:

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

Implementation

(1) For Vital Area Access

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

(2) For Equipment Qualification

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

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

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

(2) Specification of systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation.

If any of the systems listed in 'Clarification,' item 2, were excluded, explain why such systems are excluded from review;

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

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

Technical Specification Changes Required

Technical specifications will not be required.

References

NUREG-0578, Recommendation 2.1.6.b

NUREG-0660, Item II.B.2

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

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

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

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

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

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

TVA RESPONSE (Revised August 12, 1982)

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

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

Watts Bar is designed to mitigate major accidents without access to the plant outside the MCR. The postaccident sampling facility required by item II.B.3 of NUREG-0737 will be designed and shielded to allow access after an accident. It has been determined that no other additional shielding is necessary at Watts Bar.

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

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

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

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) through (3) above. Letters dated September 13, 1979 and October 30, 1979, provided clarification of staff requirements associated with items (1) through (6) above. Attachments 1 through 6 present the NRC position on these matters.

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

References

NUREG-0660, Item II.F.1

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

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

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

NRC Position

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

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

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal 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 an 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 design 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.

<u>Design Basis Maximum Range</u>

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

10^5 µCi/cc	-	Undiluted containment exhaust gases (e.g., PWR reactor building purge, PWR drywell purge through the standby gas treatment system).
	-	Undiluted PWR condenser air removal system exhaust.
10^4 µCi/cc	-	Diluted containment exhaust gased (e.g., > 10:1 dilution,as with auxiliary building exhaust air).
	-	BWR reactor building (secondary containment) exhaust air.
	-	PWR secondary containment exhaust air.
10^5 µCi/cc	-	Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).
	-	PWR steam safety valve discharge, atmospheric steam dump valve discharge.
10^2 µCi/cc	-	Other release points (e.g., radwaste buildings, fuel handling/storage buildings).
REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	(None) Sampling design criteria per ANSI N13.1.

POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/hr at 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using data detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or μCi/cc of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	Offline monitoring is acceptable for all ranges of noble gas concentrations. Inline (induct) sensors are acceptable for 10^2 μCi/cc to 10^5 μCi/cc noble gases. For less than 10^2 μCi/cc, offline monitoring is recommended. Upsteam filtration (prefiltering to remove radioactive idodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident. For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken in account in accounting for low-energy gamma radiation.

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

NRC Position

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

Changes to Previous Requirements and Guidance

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

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

(2) Specifies a shielding 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 criteria of GDC 19 of 5-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

(3) The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities.

For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ± 20%. 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 energy (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.

II.F.1, ATTACHMENT 3, CONTAINMENT HIGH-RANGE RADIATION MONITOR

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 and 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

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

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

NRC Position

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

Changes to Previous Requirements and Guidance

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

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

NRC Clarification

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

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

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

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

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

Implementation

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

Type of Review

A postimplementation review will be performed.

Documentation Required

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

Technical Specification Changes Required

Changes to technical specification will be required.

References

NUREG-0660

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

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

NRC Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWR's and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity.

Change to Previous Requirements and Guidance

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

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

NRC Clarification

(1) The containment wide-range water level indication channels shall meet the design and qualification criteria 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 of applicants for operating licenses (with an operating license date before January 1, 1982) shall be provided by January 1, 1982. Applicants with operating license dates beyond January 1, 1982, shall provide the required design information at least six months before the expected date of operation.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660

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

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

NRC Position

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

Changes to Previous Requirements and Guidance

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

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

NRC Clarification

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

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

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 (Revised August 12, 1982)

(1) NOBLE GAS EFFLUENT MONITORS

TVA will provide high range noble gas effluent monitors at Watts Bar. Due to problems involving procurement lead times, interim monitors will be provided initially. The monitor designs are described below:

A. Interim High Range Noble Gas Effluent Monitors.

Interim high range noble gas effluent monitors are at the Watts Bar plant site and will be installed by fuel load. The following notes apply to the interim noble gas monitors:

1. Each monitor consists of two General Atomic Company monitors as follows:

   a. Model RD-1 with a range of 10^{-1} to 10^{4} mR/hr

   b. Model RD-23 with a range of 10^{3} to 10^{7} mR/hr

   The model RD-1 and RD-23 monitors each utilize an ionization chamber for the detector. The RD-1 and RD-23 detectors are located at a position outside a sample pipe at which the exposure rate is approximately 10^{7}mR/hr when the sample pipe contains 10^{5} microcuries per cubic centimeter (μCi/cc) of Xe-133.

2. The monitored discharge paths are each shield building vent and the condenser vacuum pump exhaust for each unit.

3. The instrument readouts with continuous display and recording will be located in the Main Control Room (MCR).

4. Since the instrument readout will be continually available in the MCR, the required capability to obtain readings at least every 15 minutes during and following an accident will be assured.

5. The source of power for the monitors will be preferred power (vital instrument bus).

B. Permanent High Range Noble Gas Effluent Monitors

Permanent high range noble gas effluent monitors will be provided as follows:

1. For each shield building vent, a noble gas monitor with a range up to 10^{4} μCi/cc.

2. For the condenser vacuum pump exhaust for each unit, a noble gas monitor with a range up to 10^5 μCi/cc.

The above-listed monitors will comply with the requirements specified in Table II.F.1-1 in NUREG-0737. The instrument readouts with continuous display and recording will be located in the Main Control Room. The source of power will be preferred power (vital instrument bus). The permanent noble gas monitors will have ranges which overlap the ranges of existing normal range monitors by approximately a decade.

TVA expects to be able to install these permanent monitors at the first available outage of sufficient duration after initial plant startup but no later than startup after the first refueling outage.

(2) SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

For each shield building vent, collection of high-specific activity particulates and iodine on collection media for subsequent laboratory analyses will be provided.

(3) CONTAINMENT HIGH-RANGE MONITORS

Redundant high-range monitors that meet the requirements for monitoring exposure rates in containment in accordance with the requirements of Table II.F.1-3 of NUREG-0737 are provided.

A. The high-range containment monitors are General Atomic Company model RD-23. The detector of the monitor is an ionization chamber with stainless steel sealed housing.

B. The monitors have a range of 10^0 to 10^7R/hr, a uniform energy response (± 20 percent) from 80 keV to 3 MeV, and are qualified Class 1E. The monitors will operate within specifications at integrated doses up to 10^9 rads.

C. The monitors are now at the Watts Bar plant site and will be installed by fuel load.

(4) CONTAINMENT PRESSURE MONITOR

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 lb/in²g. Redundant, continuous containment pressure indication with a range up to four times the design pressure (-5 to 60 lb/in²g) 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) CONTAINMENT WATER LEVEL MONITOR

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 total ice melt, the containment water level would not exceed the 20-foot 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 existing in 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 RG 1.45.

(6) CONTAINMENT HYDROGEN MONITORS

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 remote manually actuated in the Main Control Room. The range of these monitors is from 0 to 10 percent hydrogen concentration from negative 2 lb/in^2g to positive 50 lb/in^2g pressure.

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

The hydrogen analyzers meet the applicable requirements for qualification, redundancy, 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 provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

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.

Previous guidance on the design 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 proper 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) integration 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 will 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 installation, 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 instructions in the current emergency procedures for ICC and a description of how these procedures will be modified when the final monitoring system is implemented.

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

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

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

II.F.2, ATTACHMENT 1, DESIGN AND QUALIFICATION CRITERIA FOR PRESSURIZED WATER REACTOR INCORE THERMOCOUPLES

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

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

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

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

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

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

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

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

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

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

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

(b) integration into emergency procedures,

(c) integration into operator training, and

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

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

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

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

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

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

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

TVA Response (Revised August 12, 1982)

1. Description of Proposed Final System Design

a. Additional Instrumentation

1a-1. Reactor Vessel Level Instrumentation System (RVLIS)

The RVLIS is described in Appendix A to this submittal. This information was previously submitted to the NRC by letter from L. M. Mills to A. Schwencer dated January 2, 1982. The RVLIS report is revised via this submittal to reflect the actual system design for Watts Bar. It should be noted that this information is proprietary to Westinghouse Electric Corporation, and the application for withholding this information from public disclosure (CAW-80-77) is also included in Appendix A.

1a-2. Saturation Temperature Meter (STM)

The STM will be implemented with a dual scale vertical panel meter. One scale will read actual Reactor Coolant System (RCS) pressure (0-3000 psig) and the other scale will be marked with a saturation temperature scale (212°F-695.5°F nonlinear) so that the operator can easily correlate RCS pressure to its corresponding saturation temperature. Redundant indications will be provided. The RCS hot-leg temperature indicators and the STM will be sufficiently human engineered to preclude confusion of the saturation temperature indication with the actual RCS hot-leg temperature. Individual RCS pressure and hot-leg temperature measurements are recorded on strip chart recorders; therefore, a continuous trend of saturation could be calculated if required.

b. Existing Instrumentation

1b-1. Core Exit Thermocouples

The Watts Bar Nuclear Plant core exit thermocouples (T/C) 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 Post Accident Monitoring sensors.

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

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

An example of the Watts Bar selective readings is an on-demand tabular listing of all instantaneous core exit thermocouple values.

A printout of average, instantaneous, and maximum values is provided for all thermocouple temperatures. The range will meet the suggested range of 200°F-2300°F.

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

Alarm capability is provided in conjunction with the subcooling monitor which uses the average of all the thermocouple readings in the calculations.

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

A backup analog readout is provided with the capability of selective reading of any thermocouple in the system. The present range of this system is 0-700°F. TVA has reviewed the extension of the analog readout temperature range to 2300°F, as specified in NUREG-0737. From this review, it was noted that the range could not be extended beyond 2000°F without making significant modifications to the presently installed plant hardware. TVA has concluded that a temperature range of 0-2000°F for the incore thermocouple backup analog readout represents an acceptable level of compliance with the intent of NUREG-0737 since (1) the range of the primary readout for all incore thermocouples has been extended to 2300°F, as noted above, and (2) operator actions, as presently specified in the draft inadequate core cooling (ICC) procedures from the Westinghouse Owners Group, to mitigate postulated ICC events are predicated on thermocouple temperatures significantly lower than 2000°F (approximately 1200°F). Therefore, TVA will extend the

temperature range on the backup analog thermocouple readout to 0-2000°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.

1b-2. Primary Coolant Saturation Monitoring

The plant process computer is utilized to continuously monitor pressure and temperature margins-to-saturation of the primary coolant system, and to give early warning when any of these margins reaches preset limits. The following description is provided to familiarize the operator with the methods used for this monitoring, as an aid in the evaluation of saturation data to be used for trend or display, and in recognizing the type of abnormal indication that can result when certain computer inputs are unavailable or unreliable.

Saturation pressures, temperatures (U0901 through U0905), and their associated margins-to-saturation (U0984 through U0989), are calculated based on the following primary system inputs (see Table 1b-2):

1. System pressure (lowest pressurizer or wide-range)

2. Hot leg temperature (hottest)

3. Average incore T/C temperature

4. Hottest incore T/C temperature

The single pressure input used in the calculations is selected from the lowest of the four narrow-range pressurizer pressures when system pressure is above 1700 psig. Wide-range pressure is used below 1700 psig and when there are no narrow-range inputs available and reliable. If the wide-range pressure also is unavailable or unreliable, a minimum pressure value is used (unreliable zero). This causes all margin values to be indicated as unreliable and removed from limit checking. Use of this minimum pressure in the calculations also causes the margin values to go low (usually negative), which results in a sudden zero scale pen indication for any margin being trended. Alarm printout and annunciation are suppressed under these conditions.

The hottest wide-range loop temperature is selected from the four hot leg inputs on the basis of availability and reliability also. However, only those outputs associated with this temperature are affected (pen movement, alarm suppression) if none of the four inputs are usable. (700°F unreliable is used.)

Average incore temperature and hottest incore temperature are calculated by a standard Westinghouse program. This program uses temperature values for any thermocouple that is reliable and not in alarm; scan removed status is not checked. Therefore, during normal operation, if an incore thermocouple is removed from scan, a value of zero should be entered. This will cause it to be in low alarm and not be used in the average or selected as the hottest.

A control board annunciator is actuated by means of a contact closure output from the computer when any of the six margins-to-saturation are found to be below the associated setpoint (K5502 or K5503) and will not clear until all six margins have returned to values above the setpoints. The operator can periodically test the annunciator by simply changing either of the setpoints to a value greater than a current margin value.

c. No modifications are presently planned for the instruments described in (b) above.

2. Appendix A and its references contain the necessary design analysis to support the RVLIS design described in item 1 above.

3. The RVLIS will be preoperational tested to verify proper installation, calibration, and scaling. Data will be collected during a heatup phase for verification of the functions provided in the compensation electronics. The wide-range indicators will also be marked at their appropriate percentages according to the number of reactor coolant pumps in service. Also, this test will familiarize the operations staff with the functions provided in the RVLIS.

   Due to the simplicity of the STM, we feel no additional testing of this plant feature will be required.

4. Conformance of RVLIS to Regulatory Guide 1.97, Revision 2, Category 1, requirements is shown in appendix A Table 4.1. A description of our quality assurance program is given in Topical Report TVA-TR75-1A 'Quality Assurance Program Description for Design, Construction, and Operation of TVA Nuclear Power Plants.'

   Conformance of the STM to Regulatory Guide 1.97, Revision 2, Category 1 requirements can be shown by noting that the

analog signal used to drive the STM is the RCS wide-range pressure signal. We consider the RCS wide-range pressure a Category 1 variable as defined by Regulatory Guide 1.97, Revision 2, and the design of its display will meet the Category 1 requirements of Regulatory Guide 1.97, Revision 2. As noted above, our quality assurance program is described in Topical Report TVA-TR75-1A.

The Primary Coolant Saturation Monitoring Program's inputs conform to the Category 1 requirements of Regulatory Guide 1.97 with the exception of the core exit thermocouple inputs and only one wide range RCS pressure measurement is used (see Table 1b-2).

The core exit thermocouple system does not conform to the requirements of attachment 1 of NUREG-0737 item II.F.2. The primary and backup displays for the core exit thermocouples are not separated in accordance with Regulatory Guide 1.75. Presently all the thermocouple cables are routed together from the reactor vessel head to the main control room. The thermocouples are hardwired to the backup display directly, and a parallel connection is made at the input terminals of the backup display for connection of the thermocouples to the primary display. The thermocouples are normally isolated from the primary display by a series of two normally open contacts in the computer input racks. The backup display is not qualified to the requirements of Regulatory Guide 1.97 for Category 1 circuits. The reference junctions located inside containment which provide a controlled 160°F reference temperature are not qualified to the Category 1 requirements of Regulatory Guide 1.97, but the reference junctions would not have to function in the post-LOCA environment. The only requirement of them would be to provide a transition from thermocouple extension wire to copper signal cable (i.e., terminal block). This deviation is justifiable because the primary display can compensate its measurement for higher reference temperatures (i.e., greater than 160°F) or lower reference temperatures (i.e., less than 160°F) through use of resistance temperature detectors (RTDs) imbedded in the terminal blocks. Also the RTDs could be monitored at the computer input racks and used to compensate the backup display, if necessary, should the primary display fail. The reference junctions are located in the incore instrument room and are seismically mounted. The thermocouple extension wire which connects the thermocouples to the reference junctions is not qualified for post-LOCA service, but the cables were specified for use in 600°F ambient temperature environment and were tested for accuracy at 400°F. The specified accuracy of $\pm$ 2°F at 400°F ambient was met in the test. The thermocouple extension cables will not be exposed to caustic spray (i.e., containment spray) since they are routed in iron conduit. The copper signal cable used to connect the reference junctions to the backup display is qualified for post-LOCA service. The connectors are not qualified per the guidelines of Regulatory Guide 1.97 and will be replaced when a more reliable connector is developed.

5. The plant computer at Watts Bar is a P-2500 system supplied by Westinghouse. The computer's function in ICC monitoring is to provide a primary display for the core exit thermocouple system, calculate a margin to saturation as described in item 1b-2 and display the margin to saturation calculations. No availability data on the Watts Bar computer has been submitted to us by Westinghouse at this time, but a 2,900 hour test conducted on the Sequoyah plant computer (P-250) which is very similar to the Watts Bar plant computer has shown an availability of 99.95 percent for the Sequoyah system.

6. It is presently anticipated that all ICC monitors for Watts Bar will be operational before initial criticality of each unit.

7. TVA is working as a member of the Westinghouse Owners Group (WOG) on developing guidelines for the use of ICC instrumentation. The guidelines, and a discussion of the analysis used to develop the guidelines will be provided to the NRC before fuel loading.

8. As discussed in item 7 above, the WOG is developing guidelines for the use of ICC instrumentation. The operator action instructions in the current Emergency Operating Instructions (EOI) covering use of existing ICC instrumentation will be modified based on the WOG guidelines and the additional ICC instrumentation. A summary of the key operator action instructions in the final EOIs covering use of IC instrumentation will be provided to the NRC before fuel loading.

9. The documentation discussed in items 7 and 8 above will be provided to the NRC before fuel loading.

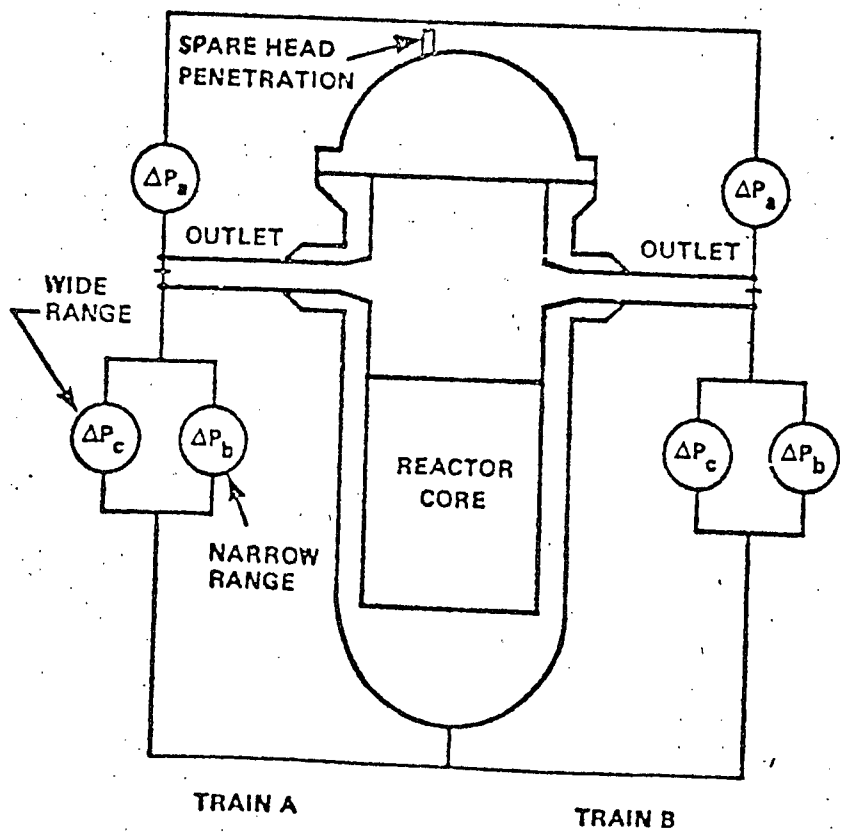
TABLE 1b-2

PROGRAM INPUTS

AP Symbol	Instrument		Range	Units
P0480A	PT-68-340	Pressurizer 1 Press	1700-2500	lb/in²g
P0481A	PT-68-334	Pressurizer 2 Press	1700-2500	'
P0482A	PT-68-323	Pressurizer 3 Press	1700-2500	'
P0483A	PT-68-322	Pressurizer 4 Press	1700-2500	'
P0499A	PT-68-68A	RCL System Press	0-3000	'
T0419A	TE-68-1	RCLA Hot Temp	0-700	degree F
T0439A	TE-68-24	RCLB Hot Temp	0-700	'
T0459A	TE-68-43	RCLC Hot Temp	0-700	'
T0479A	TE-68-65	RCLD Hot Temp	0-700	'
U0090		Inst Value of Hottest Incore T/C	0-2500	'
U0091		Inst Value of Average Incore T/C	0-2500	'
K5502		PSAT Margin Annunc Alarm Setpt	200	lb/in²
K5503		PSAT Margin Annunc Alarm Setpt	15	degree F

PROGRAM OUTPUTS

U0900	Press Value Used for Sat Calc	lb/in²g
U0901	Sat Press for Hottest RCL Temp	'
U0902	Sat Press for Hottest T/C	'
U0903	Sat Press for Avg Incore Temp	'
U0904	Hot Leg Temp Used for Sat Calc	degree F
U0905	Sat Temp for System Pressure	'
U0984	PSAT Margin for Hottest RCL Temp	lb/in²
U0985	PSAT Margin for Hottest T/C	'
U0986	PSAT Margin for Avg Incore Temp	'
U0987	TSAT Margin (RCL T) for Sys Press	degree F
U0988	TSAT Margin (U0090) for Sys Press	'
U0989	TSAT Margin (U0091) for Sys Press	'



Reactor Vessel Level Instrument System

Figure II.F.2-1

APPENDIX A

TO

INADEQUATE CORE COOLING MONITORS

WATTS BAR NUCLEAR PLANT

RESPONSE TO NUREG-0737

ITEM II.F.2