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

October 17, 1979

Mr. D. G. Eisenhut, Acting Director
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

Enclosed is our response to your letter dated October 13, 1979, to all operating nuclear plant owners in which you requested commitments to comply with the requirements of NUREG-0578 as outlined in the letter. TVA is a member of the BWR Owners' Group and we have actively participated in the preparation of the Group's generic positions on the NUREG-0578 items which were submitted to Mr. Denton in an October 17, 1979, letter from T. D. Keenan, Group Chairman. While this response is specific to Browns Ferry Nuclear Plant, it is compatible with the generic BWR positions.

Complying with some of the NUREG-0578 items may require changes in Browns Ferry's technical specifications or raise an unreviewed safety question. Pursuant to 10 CFR 50.59, TVA would then be required to submit an application for a license amendment and receive NRC approval before the required modifications could be made. As discussed by the staff in recent regional meetings, the severe time constraints for implementing changes to meet NUREG-0578 commitments restrict the NRC premodification review time to a minimum.

Although TVA will perform the evaluation required by 10 CFR 50.59, it will not be possible to obtain specific NRC approval for several license amendments if the implementation is to be completed in accordance with the schedules in the enclosure. We, therefore, request the NRC to provide written notification that approval of the license amendment before completion of the change will not be required for those items specifically related to the NUREG-0578 commitments enclosed.

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Mr. D. G. Eisenhut October 17, 1979

In addition, implementation of some of the NUREG-0578 items will cause unscheduled outages. For example, on item 2.1.3.a, Direct Indication of Valve Position, we have indicated that plant modifications are planned for November 1979 for unit 3, January 1980 for unit 1, and March 1980 for unit 2. For unit 2, this effort would require a one-week (minimum) unscheduled outage in March 1980 which would result in major expenditures for replacement power. Based on this impact on power system costs and the fact that TVA does not consider this to be a significant safety issue for BWR's, we request a delay in implementing this modification until the next regularly scheduled refueling outage for unit 2 which will begin on September 1, 1980.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure

1202 082

2.1.1 Emergency Power Supplies

NRC Position

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

Pressurizer Heater Power Supply

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

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

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

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

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

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

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

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

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

TVA Response

Pressurizer Heater Power Supply

This position is not directly applicable to Browns Ferry; boiling water reactors do not use a pressurizer system to establish and maintain primary system pressure.

Natural circulation in the BWR, as discussed in NEDO-24708, is strong and inherent in all off-normal modes of operation as long as sufficient vessel inventory is maintained, and natural circulation is independent of any powered system. This is because, even in normal operation, the BWR is essentially an augmented natural circulation machine. Because the BWR operates in all modes with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. Thus, there is no need for emergency power to maintain natural circulation or to keep the system pressurized.

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

All main steam line relief valves that are part of the automatic depressurization system are supplied with safety grade, diverse power.

BWR's do not have block valves.

The reactor vessel level indication instrument channels for safety system activation and control are already powered by emergency power.

For the reasons stated above, there is no need for action in response to NRC position 2.1.1.

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2.1.2 Performance Testing for Relief and Safety Valves

NRC Position

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

TVA Response

TVA is in agreement with the consensus of the BWR Owners Group that additional testing of safety and relief valves is not required. The basis for our position is presented in the BWR Owners Group submittal on this item. The conclusions of that submittal are summarized as follows:

1. S/RV's are routinely tested for the expected mode of operation (saturated steam) by inplace functional tests and by frequent usage in mitigating plant transients.

2. Two-phase flow at high pressure is not calculated to occur for design-basis transient and accidents.

3. Inadvertent passage of two-phase flow is unlikely since feedwater and high-pressure injection systems are automatically tripped on high reactor vessel level. TVA will further investigate the present qualification of the high level trip systems to determine if upgrading is warranted.

4. In the three events discussed in the BWR Owners Group submittal that two-phase flow did occur (prior to installation of high level trips), the S/RV's reclosed.

5. TVA in association with General Electric Company has pursued an active course in improving S/RV performance and reliability, both by valve modifications and by system improvements to reduce the number of challenges to the S/RV's.

2.1.3.a Director Indication of Valve Position

NRC Position

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

TVA Response

TVA will provide main control room indication of valve position of the safety/relief valves by use of a downstream acoustic monitoring system. The system will consist of an accelerometer on the discharge piping of SRV's (one per SRV) that will indicate if a valve is open. The system will be qualified for a LOCA environment, will be powered from Class 1E power supply, and will be seismically qualified to preclude interaction with safety-related equipment but will not be qualified to function during or after a seismic event.

The monitoring system will not be available for all Browns Ferry units by the requested implementation date since delivery cannot be made by that date. Based on our present equipment delivery schedule, equipment installation is expected to begin November 1979 for unit 3, January 1980 for unit 1, and March 1980 for unit 2.

Until the acoustic monitoring system is in service, individual valve position indication will be determined by temperature sensors located in the discharge piping downstream of each valve. These sensors display individual valve tailpipe temperature and initiate a high temperature alarm in the control room.

2.1.3.b Instrumentation for Inadequate Core Cooling

NRC Position

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. 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.

TVA Response

GE has been examining this requirement on a generic basis for the BWR Operating Reactor Owners Group. The present schedule for issue of this information from GE is November 1979. However, this schedule is being coordinated with and is at the discretion of the Bulletins and Orders Task Force (as discussed in the BWR Owners Group's response to NUREG 0578, item 2.1.9).

Based upon the November schedule, TVA will develop procedures, provide a description of the existing instrumentation, and implement training for the operators to use to recognize inadequate core cooling conditions within four months after review completion.

Thus far, a need for additional instrumentation has not been identified. However, TVA will continue to work with GE to determine if additional instrumentation is required at Browns Ferry. TVA will make every effort to meet the required implementation dates of January 1980 for identification of this instrumentation and January 1981 for installation.

2.1.4 Containment Isolation Provisions for PWR's and BWR's

NRC Position

All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

All plants shall give careful reconsideration to the definition of essential and nonessential systems, shall identify each system determined to be essential, shall identify each system determined to be nonessential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to NRC.

All nonessential system shall be automatically isolated by the containment isolation signal.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliverate operator action.

TVA Response

Browns Ferry either complies or will comply with the NRC positions on the four portions of this item in the following manner:

1. SRP 6.2.4 requires diversity in parameters used for containment isolation. Browns Ferry complies by isolating on (a) low reactor water level and (b) high drywell pressure.

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

3. Browns Ferry design complies with NRC requirements on the automatic isolation of nonessential systems. Any changes necessary as a result of study in item 2 will be made.

4. Browns Ferry design will comply with the NRC's requirements by requiring manual actions on the controls of individual components to change their status after the containment isolation signal has been cleared.

The design studies in item 2 described above will be completed by January 1, 1980, and identified design changes made expeditiously. TVA expects to begin implementing all required design changes identified in item 4 above by spring 1980. This ambitious schedule assumes that TVA will experience no difficulty in procuring required components. In the interim, procedural controls will be incorporated as necessary to prevent automatic change of status of isolation valves after the containment isolation signal has been cleared.

2.1.5.a Dedicated Hydrogen Control Penetrations

NRC Position

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

TVA Response

Post-accident hydrogen control for Browns Ferry is provided by inerting the primary containment during normal operation. After an accident, long-term combustible gas concentrations are maintained by the containment atmospheric dilution (CAD) system. This system is designed to purge small quantities of the containment atmosphere to the standby gas treatment system while adding makeup nitrogen to the containment. The CAD system meets NRC regulations on redundancy and single failure criteria. The system is designed to limit the maximum opening from the containment to two inches even in the event of the worst single failure during CAD system operation.

For the reason stated above, there is no need for action in response to NRC position 2.1.5.a.

.5.b Inerting BWR Containments

Position

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWR's.

TVA Response

Browns Ferry has an inerted containment and maintains inerted conditions per technical specification requirements.

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2.1.5.c Installing Hydrogen Recombiners in LWR's

NRC Position

All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

TVA Response

This requirements is not applicable to Browns Ferry. The Browns Ferry design utilizes inerted containment design for combustible gas control.

2.1.6.a Systems Integrity for High Radioactivity

NRC Position

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

    a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

    b. Measure actual leakage rates with system in operation and report them to NRC.

2. Continuing Leak Reduction

    Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

TVA Response

TVA will investigate practical leakage reduction measures on systems which may contain radioactive fluids post-LOCA and will examine such systems as the reactor core isolation cooling system, high-pressure coolant injection system, core spray system, residual heat removal system, and waste disposal system.

This examination will include a study of valve steam packing leakoffs, rotating seals on equipment, gasketed connections or joints, drains piped to open connections, and reactor drainage system.

TVA will identify which of the above systems from which leakage may be measured and will report measured leakage from the above systems to NRC.

TVA will identify the above systems that may be leak checked and will implement a periodic leak inspection program on these systems.

The above investigations will be completed by January 1, 1980.

2.1.6.b Plant Shielding Review

NRC Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, 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 post-accident 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, post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

TVA Response

TVA plants are specifically designed to mitigate major design basis events with no access outside the MCR being required. With this goal in mind, the plants were not specifically designed for any access outside the main control room. To specifically design for guaranteed access at anytime in most parts of the reactor building is not feasible. However, the current designs may 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 hs provided some shielding for post-accident access. TVA will make design changes in shielding if evaluations identify feasible modifications which should significantly enhance desirable access. The guidelines for the evaluations are given below.

TVA will assume a TID 14844 radioactivity release into the primary containment. A summation of the radioactivity levels from sump water leakage from process systems in the reactor building will be made. The next step will be to calculate the source terms for the suppression pool recirculating piping, pumps, and valves installed in the reactor building assuming that a TID 14844 release had occurred. TVA will then identify the vital areas in the reactor building which may need to be entered for servicing during an accident recovery period. The shielding in these vital areas will be reevaluated to assess its effectiveness in such a circumstance. The occupancy time limits, taking into consideration transit time, airborne radioactivity levels, and gamma shine intensities, will then be calculated for the vital reactor building areas.

2.1.7.a Auto Initiation of Auxiliary Feedwater

NRC Position

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

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

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

3. Testability of the initiating signals and circuits shall be a feature of the design.

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

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

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

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

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

TVA Response

Although Browns Ferry does not have an auxiliary feedwater system, portions of the Browns Ferry ECCS network perform comparable functions. These ECCS systems are safety grade and meet the intent of the above stated position.

2.1.7.b Auxiliary Feedwater Flow Indication

NRC Position

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented.

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

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

TVA Response

As noted in the response to 2.1.7.a, portions of the Browns Ferry ECCS network serve similar functions for post-accident recovery as the auxiliary feedwater system for PWR's. The flow indications for these systems are safety grade and meet the intent of the above position.

2.1.8.a Improved Post-Accident Sampling Capability

NRC Position

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

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

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

TVA Response

A design and operational review of the reactor coolant sampling systems and analysis facilities is being performed and will be complete by January 1, 1980. TVA expects to complete required modifications by January 1, 1981, provided that equipment procurement/installation conflicts are not encountered. These modifications will make provisions for sampling water from the reactor coolant system for the degraded accident condition. TVA will also identify the type and nature of onsite analysis required. If practical, TVA will procure the required analysis equipment and locate, design and build an onsite analysis facility.

Until the design modifications are complete, procedures will be devised to evaluate the primary coolant system activity depending on the accessibility of the sampling stations for particular degraded conditions.

2.1.8.b Increased Range of Radiation Monitors

NRC Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near term.

1. 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 to be necessary to cover the ranges of interest.

   a. 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.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10^{-7} μCi/cc (Xe-133) 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 shall overlap by a factor of ten.

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

3. 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 designed and qualified to function in an accident environment.

TVA Response

TVA will provide redundant safety high-range noble gas effluent monitors.

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

TVA will provide redundant safety grade radiation monitors to meet NRC's high-range monitor requirements. These monitors will be isolated from the containment atmosphere yet not located outside the containment shielding. Exposure rates at selected detector location from activity within the containment will be high enough such that instrument readings can be correlated with containment activity throughout the course of the accident.

In the interim, procedures will be developed as practical to estimate release rates if existing effluent instrumentation goes off scale.

2.1.8.c Improved Inplant Iodine Instrumentation

NRC Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

TVA Response

Browns Ferry has portable low-volume air samplers, each equipped with a particulate filter followed by a charcoal adsorber to collect iodine isotopes. The particulate filter will be counted in the health physics laboratory for gross activity and the charcoal adsorber sent to the radiochemical laboratory for a gamma isotopic analysis for radioactive iodines. If necessary, as necessitated by a high-gross activity, the particulate filter will also be sent to the radiochemical laboratory for an isotopic analysis. The primary difference in obtaining inplant airborne isotopic concentrations for accident and routine operating conditions is the time required for sampling. A shorter sample time could be necessary for accident conditions because of the presence of high isotopic concentrations.

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

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

2.1.9 Transient and Accident Analyses

NRC Position

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of Some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required--LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the anlytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy.

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these anlayses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCA's, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NASS vendors be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons, together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to ensure the adequacy of the analytical methods being used to generate emergency procedures.

TVA Response

TVA is pursuing the required analyses and the development of new procedures and training guidelines with other utilties through the BWR Owners Group.

Small break loss-of-coolant accident analyses have been performed and issued in NEDO-24708. The report presents a comprehensive study of BWR system response to small breaks as well as GE recommendations for operator guidelines for these accidents. These analyses and guidelines were developed in response to the July 13, 1979, question set of the Bulletins and Orders Task Force (B&OTF). TVA will begin incorporating these analyses and guidelines into operator procedures and training programs, and implement training within four months after the B&OTF review is complete.

Inadequate core cooling analyses and operator guidelines have not been completed. These analyses are also being developed in response to the July 13, 1979, B&OTF question set. The schedule for this work is being coordinated between GE and the B&OTF. The present schedule for completion and issue of these items is November 1979. TVA plans to have these analyses and guidelines incorporated into emergency operating procedures and training programs, and complete operating training within four months of the completed review.

Analyses and guidelines for emergency procedures to improve operator performance during transients and accidents are discussed in the BWR Owners Group reponse to NUREG-0578, item 2.1.9.

TVA will incorporate these analyses and guidelines in emergency procedures and training programs within four months after the guidelines are established as required by NUREG-0578.

2.1.9 Enclosure 3 Position (1) Containment Pressure
(2) Hydrogen Monitoring
(3) Water Level Monitoring

NRC Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented.

1. A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

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

3. 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. Also for PWR's, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000-gallon capacity. For BWR's, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment pressure, hydrogen concentration, and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testibility. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

TVA Response

Additional Item 1 - Containment Pressure Monitoring

TVA will determine if the present containment pressure indication meets or can be modified to meet the required pressure range from -5 psig to four times the design pressure of the containment vessel and if the equipment is qualified for this range. If not, TVA will provide the appropriately qualified equipment.

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Additional Item 2 - Hydrogen Concentration Monitoring

TVA is planning to install a new hydrogen concentration monitoring system at Browns Ferry. TVA will determine if the planned system meets or can be modified to meet ranges for hydrogen detection and operating pressures and if it is qualified for these ranges. If not, TVA will provide the appropriately qualified equipment.

Additional Item 3 - Containment Water Level Monitoring

TVA will determine if the current design for suppression pool level indication meets or can be modified to meet the requirement to cover the range from the lowest ECCS suction point (ring header) to 5 feet above the normal water level of the suppression pool and if the design is qualified for this range. If not, TVA will provide the appropriately qualified equipment.

2.1.9 Position (4) Reactor Coolant System Venting

NRC Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. 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. In particular, these vents shall be safety grade and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating vent usage shall be discussed.

TVA Response

TVA concurs in the consensus of the BWR Owners Group that the Browns Ferry existing reactor venting capability is fully satisfactory. The justification for our position was presented to NRC during the October 11, 1979, topical meeting and is also contained in the BWR Owners Group submittal in response to NUREG-0578. The existing Browns Ferry reactor venting system is represented accurately in the BWR Owners Group submittal. Additional considerations will be investigated consistent with those determined by the BWR Owners Group.

2.2.1.a Shift Supervisor's Responsibilities

NRC Position

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

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

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

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

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

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

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

TVA Response

1. TVA's administrative procedures, shift supervisor job descriptions, and training programs emphasize the primary management responsibility of the shift supervisor. In addition, periodic retraining acts to reinforce his command responsibilities. While these existing measures provide a high level of confidence that the shift supervisor has primary management responsibility for safe operation of the plant, TVA will periodically issue a management directive which emphasizes this assignment of responsibility.

2a. Plant administrative procedures have been reviewed to ensure that they clearly define the authority and responsibilities of each position on shift. The duties and responsibilities of the shift supervisor, as specified in the job description, are consistent with position statement 2a.

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

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

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

2c. In the event that the assistant shift engineer (shift supervisor) is absent, the unit operator will be the lead operator on the unit to which he is assigned. An additional licensed operator will be available in the control complex to act as an assistant to the unit operator in abnormal or emergency situations.

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

4. The administrative duties of the shift supervisor will be reviewed by the senior officer of TVA responsible for plant operations. Administrative functions that detract from or are subordinate to ensuring safe operation of the plant will be assigned to other employees. A clerk has been assigned to the shift engineer's office on each shift to perform administrative details formerly done by the shift engineer.

2.2.1.b Shift Technical Advisor

NRC Position

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

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

TVA Response

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

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

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

On each shift, there will be one shift technical advisor, However, this person will be assigned other duties when his duties as shift technical advisor are not required, provided that his availability is not compromised. TVA is optimistic that a substantial portion of the Shift Technical Advisor training may be completed by January 1, 1981.

As an interim policy by January 1, 1980, (1) an additional SRO will be placed on each shift to act as Shift Technical Advisor as circumstances require, and a duty engineer shall also be designated on call for advice in support of the shift technical advisor, or (2) a plant experienced degreed engineer will be placed on shift to act as shift technical advisor.

TVA believes that a multi-disciplined review group is necessary to adequately investigate LER's. TVA's Nuclear Experience Review Panel presently reviews all licensee event reports. When applicable, results of the review will be incorporated in TVA's operator training and requalification programs. In addition, periodic training sessions are conducted for each shift crew. The material covered during these sessions include, but is not limited to, licensee event reports, operator errors, recent equipment problems, changes to technical specifications, and general plant status. The Shift Technical Advisors shall have additional responsibilities in being cognizant of the results of the LER review as applied to Browns Ferry.

2.2.1.c Shift and Relief Turnover Procedures

NRC Position

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

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

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

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

   c. Identification of systems and components that are in a degraded mode of operation permitted by the technical specifications. For such systems and components, the length of time in the degraded mode shall be compared with the technical specifications action statement (this shall be recorded as a separate entry on the checklist).

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

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

TVA Response

TVA will develop and implement shift and relief turnover procedures that will provide assurance that the oncoming shift possesses adequate knowledge of critical plant status information and system availability. A checklist or similar hard copy will be completed by offgoing and oncoming shifts at each shift turnover.

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

2.2.2.a Control Room Access

NRC Position

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

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

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

TVA Response

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

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

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2.2.2.b Onsite Technical Support Center

NRC Position

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the technical support center under emergency conditons. These documents shall include, but not be limited to, general arrangement drawings, P&ID's, piping system isometrics, electrical schematics, and photographs of components installed without layer specifications (e.g., field-run piping and instrument tubing).

TVA Response

An onsite technical support center will be established. This support center will meet the same habitability requirements of the main control room and will contain reliable communications equipment. A complete set of functional plant drawings and necessary technical information will be placed in or near the support center. Prior to January 1, 1981, equipment will be installed in the support center to improve the plant monitoring capability of technical support personnel.

The plant Radiological Emergency Plan will be amended to establish the technical support center and specify the personnel who will staff it in the event of an emergency.

2.2.2.c Onsite Operational Support Center

NRC Position

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

TVA Response

An operational support center with communications to the main control room will be established. The plant Radiological Emergency Plan will be amended to establish this center and to specify the personnel who will report to this center in the event of an emergency.

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2.?.3 Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability

NRC Position

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures, and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.

2. Within 24 hours, bring the plant to cold shutdown.

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee.

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

TVA Response

No action is required by TVA at this time.

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Near Term Emergency Preparedness Improvements Implementation Schedule

NRC Position

As stated in enclosure 7 and 8 of subject letter.

TVA Position

Item #1: TVA's Radiological Emergency Plan (REP) will be upgraded taking Regulatory Guide 1.101 requirements into consideration. This upgrading will include uniform action level criteria and will be completed about July 1, 1980.

Item #2: The short-term items identified by the Lessons Learned task force will be included in the action level criteria as applicable.

Item #3: An emergency operations center for Federal, State, and local officials will be identified in the Browns Ferry region, and communications with the TVA emergency communications system will be established prior to July 1, 1980. The inplant technical support center planning is underway and will be complete prior to January 1, 1981.

Item #4: Additional thermoluminescent dosimeters have been ordered for monitoring offsite environment around Browns Ferry. This additional offsite monitoring will be implemented by mid 1980.

Item #5: The Alabama State and local REP have been concurred in by NRC and consequently meet current criteria. TVA is working with the State to evaluate the need to develop appropriate emergency action to a distance of ten miles and will work with the State in support of upgraded NRC criteria. While TVA will work toward a January 1, 1981, completion date, the State has the authority and the responsibility for developing this plan in accordance with their prerequisites.

Item #6: TVA currently tests the Browns Ferry REP on an annual basis. TVA will continue to test both the plant and State plans on this basis. In addition, the revised plans will be tested in a major drill including Federal, State, local, and TVA emergency organizations within the next five years.

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