TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 31, 1979

Mr. Domenic Vassallo, Acting Director
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Vassallo:

In the Matter of the Application of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

Enclosed are ten copies of TVA's revised responses to NUREG 0578, Lessons Learned Requirements, for the Sequoyah Nuclear Plant. Our responses fully address the NRC's letter (To All Operating Nuclear Power Plants) from Harold Denton. Mr. Denton's letter provided additional guidance on lessons learned short term requirements.

For your convenience, the responses are presented in binders with dividers separating each item. Each item includes the NRC position statement with the associated TVA response.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure (10)

1296 196

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EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Heaters

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.

CLARIFICATION

1. In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply.

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

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situtation.
   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.
   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

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

EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Heaters

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA policy on pressurizer heaters is consistent with the NRC position. The ability of the Sequoyah Nuclear Plant design to meet each of the NRC recommendations is addressed in the following response.

RESPONSE

The SQN pressurizer heaters are powered and controlled from Class IE sources (see FSAR figures 8.3-10, 8.3-11, 8.3-12, and 8.3-13). The motive and control power interfaces with the emergency buses are qualified in accordance with safety-grade requirements. All four heater banks will trip on a Safety Injection signal when in the normal mode. After safety injection reset and level recovery in the pressurizer, one backup heater bank (1c) would operate automatically. The other two backup heater banks and the control bank would not come on automatically but are manually activated. In the event of a loss of offsite power and safety injection signal, two backup heater banks rated at 485 KW each can be manually activated by hand switches in the main control room, 90 seconds after emergency power becomes available. The required operator actions are specified in the Sequoyah Emergency Operating Instructions (EOI 5).

CLARIFICATION ITEMS

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

Power Train	Heater Bank
1A-A	1A-A
1B-B	1c (automatic)
2A-A	2A-A
2B-B	2c

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

1296 199

CLARIFICATION ITEMS (continued)

3. The Sequoyah design provides for the required power for each bank of pressurizer heaters.

4. As specified in the above response, this cability is available at Sequoyah.

5. The existing Sequoyah Emergency Operating Instructions (EOI) acount for the considerations specified in this clarification item.

6. The above response specifies that Class 1E interfaces for main power and control power are protected by safety-grade circuit breakers.

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

1296 200

Emergency Power Supply (2.1.1)

Pressurizer Level and Relief Block Valves

POSITION

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

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

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

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

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

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

CLARIFICATION

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any changover of the PORV and block valve motive and control power from the normal offsite power to the emergency offsite power is to be accomplished manually in the control room.

1296 201

4. For those designs where instrument air is needed for operation, appropriate electrical power must also be applied via the emergency power sources.

Emergency Power Supply (2.1.1)

Pressurizer Level and Relief Block Valves

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

The design for Sequoyah Nuclear Plant is consistent with the NRC positions concerning power supply for pressurizer relief and block valves and pressurizer level indication.

RESPONSE

The power-operated relief valves (PORV) and their associated block valves and control components are classified as Class 1E and are supplied from the emergency onsite power supply if offsite power is lost. The relief valves and their associated block valves are powered from opposite power trains. All connections to the emergency power supply are through devices that are qualified in accordance with safety grade requirements. For a description of the PORV and block valves, see FSAR Sections 5.1 and 5.2.2 and figure 5.1-6.

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

CLARIFICATION ITEMS

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

2. The redundancy built into the Sequoyah systems provides for independent emergency power for PORV and block valves.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to emergency power is accomplished manually in the control room as specified in the Sequoyah Emergency Operating Instructions.

4. Where instrument air is needed for operation, any associated electrical power required can be supplied by emergency power sources.

1296 203

PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES (2.1.2)

POSITION

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

CLARIFICATION

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70.

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

1296 204

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981.

1296 205

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA,s commitment to a program for testing of relief and safety valves is presented below. This program is consistent with the NRC position.

RESPONSE

TVA is actively pursuing a joint effort with other members of the utility industry which will develop requirements for a generic test facility and program for reactor coolant system relief and safety valve prototypical testing. This joint effort will identify expected valve operating conditions through analytical studies and through these bounding analyses develop performance specifications for the test facility.

TVA will submit to NRC a description of and schedule for the generic performance testing of these valves as soon as this is available.

Upon completion of sufficient analysis to identify the environmental conditions which may exist, TVA will provide associated control circuits, piping, and supports which are qualified for such an environment.

CLARIFICATION ITEMS

1. See paragraph 1 of the above response.
2. Testing will demonstrate valve operability under various flow conditions.
3. See paragraph 1 of the above response.
4. Test conditions will include the effect of piping on valve operability.
5. The test results will provide data that would permit an evaluation of discharge piping and supports for those components not tested directly.
6. See paragraph 2 of the above response.
7. The testing is expected to be complete by July 1, 1981.

1296 206

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3a)

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.

CLARIFICATION

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

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

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

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualificiation.

1296 207

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided.

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3a)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Position indication in the main control room for power operated relief valves is currently available at Sequoyah. TVA will provide main control room indication of valve position of the pressurizer safety valves as specified in the following response.

RESPONSE

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

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

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

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

An environmentally qualified acoustic monitoring system for the three safety relief valves on each unit will be provided. An accelerometer will be mounted on the valve discharge line just downstream of each valve. The accelerometer signals will go to a charge converter inside containment which will be mounted in a NEMA-4 enclosure. A valve flow indicator module will be located in the main control room. The flow indicator module will give positive indication of the fully open and fully closed position of each valve. An alarm in the main control room will indicate when any valve is not in the fully closed position.

1296 209

DIRECT INDICATION OF POWER-OPERATED
VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3a)

CLARIFICATION ITEMS

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

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

3,4,5. Valve position indication for Sequoyah Nuclear Plant will meet seismic and environmental qualification requirements as specified for Sequoyah.

1296 210

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

SUBCOOLING METER

POSITION

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 NUREG-0578)

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.

CLARIFICATION

1. The analysis and procedures addressed in paragraph one above will reviewed and should be submitted to the NRC "Bulletins and Orders Task Force" for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

1296 211

6. Each PWR should have: (A.) Safety grade calculational devices and display (minimum of two meters) or (B.) a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

INFORMATION REQUIRED ON THE SUBCOOING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.) ____________

Display Type (Analog, Digital, CRT) ____________

Continuous or on Demand ____________

Single or Redundant Display ____________

Location of Display ____________

Alarms (include setpoints) ____________

Overall uncertainty (°F, PSI) ____________

Range of Display ____________

Qualifications (seismic, environmental, IEEE279) ____________

Calculator

Type (process computer, dedicated digital or analog calc.) ____________

If process computer is used, specify availability. (% of time) ____________

Single or redundant calculators ____________

Selection Logic (highest T., lowest press) ____________

Qualifications (seismic, environmental, IEEE279) ____________

Calculational Technique (Steam Tables, Functional Fit, ranges) ____________

Input

Temperature (RTD's or T/C's) ____________

Temperature (number of sensors and locations) ____________

Range of temperature sensors ____________

Uncertainty* of temperature sensors (°F at 1σ) __________

Qualifications (seismic, environmental, IEEE279) __________

Pressure (specify instrument used) __________

Pressure (number of sensors and locations) __________

Range of Pressure sensors __________

Uncertainty* of pressure sensors (PSI at 1σ) __________

Qualifications (seismic, environmental, IEEE279) __________

<u>Backup Capability</u>

Availability of Temp & Press __________

Availability of Steam Tables etc. __________

Training of operators __________

Procedures __________

*Uncertainties must address conditions of forced flow and natural circulation

1296 214

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

SUBCOOLING METER

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA will provide continuous monitoring of the deviation from saturation conditions. The plant computer will be used to perform this function. Procedures are being developed which will be used by the operator to recognize inadequate core cooling with currently available instrumentation. Operator instruction for primary coolant saturation indication will emphasize the need to use related plant parameters.

RESPONSE

TVA will provide continuous monitoring of the deviation from saturation conditions. This saturation readout will utilize output. The plant computer will be used to perform this function.

The plant computers presently monitor reactor system hot leg temperatures and pressurizer pressure. In addition, steam table conversion routines are a part of the computer software. Programs will be added to calculate saturation temperature corresponding to the measured pressurizer pressure. In the event any hot leg temperature measurement approaches the saturation temperature by a predetermined amount, an alarm will occur in the control room. The operator will be able to observe the saturation pressure and system pressure and compare the two by trending them on computer output recorder in the main control room.

TVA is developing procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation.

CLARIFICATION ITEMS

1. The guidelines for procedures specified in the above response are being developed by the Westinghouse Owners' Group in response to the Bulletins and Orders task force. TVA will provide plant procedures based on these guidelines.

2. TVA recognizes that a continuous monitoring of margin to saturation conditions will be provided.

3. Redundant safety grade temperature input from each hot leg and/or multiple core exit thermocouples are provided for measurement of saturation conditions.

4. Redundant safety grade system measurement is provided at Sequoyah.

CLARIFICATION ITEMS (cont.)

5. Continuous monitoring of the primary coolant saturation conditions will be provided as specified in the above response.

6. The margin to saturation can be continuously displaced on a trend recorder in the main control room. The backup trend recorder is available. Saturation curves are provided in the main control room and procedures will require the use of these curves on the loss of indication of margin to saturation. We expect the computer availability to exceed 99 percent.

7. Required instrumentation will be upgraded to the applicable requirements of Regulatory Guide 1.97 when they are fully developed. We consider the present design we are pursuing to be adequate.

8. Changes to the Sequoyah design will not effect the reactor protection or engineered safety features systems.

1296 216

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

POSITION

Licensees shall provide a decription 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.

CLARIFICATION

1. Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements to or a synthesis of existing measurements which meet safety-grade criteria.

2. The evaluation is to include reactor water level indication.

3. A commitment to provide the necessary analysis and to study advantages of various instruments to monitor water level and core cooling is required in the response to the September 13, 1979 letter.

4. The indication of inadequate core cooling 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).

   b) it must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

1296 217

5. The indication must give advanced warning of the approach of inadequate core cooling.

6. The indication must cover the full range from normal operation to complete core uncovering. For example, if water level is chosen as the unambiguous indication, then the range of the instrument (or instruments) must cover the full range from normal water level to the bottom of the core .

1296 218

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Analysis and procedures for the detection of inadequate core cooling using existing instrumentation are currently being developed in conjunction with the Westinghouse Owners' Group. This will be the primary method for detecting inadequate core cooling. In addition, TVA will provide instrumentation to measure water level in the reactor vessel down to the bottom of the hot leg piping.

RESPONSE

Analysis and procedures for the detection of inadequate core cooling using existing instrumentation are currently being developed in conjunction with the Westinghouse Owners' Group

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

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

1296 219

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

RESPONSE (cont.)

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

To provide the required accuracy for water level measurement, temperature measurements of the reference legs are provided. These measurements together with the reactor coolant temperature measurements are used to compensate the differential pressure transducer outputs for differences in reference leg temperature, particularly during the environment inside the containment structure following an accident.

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

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

RESPONSE (cont.)

Instrumentation for the operator for the Reactor Vessel Level Instrumentation System is intended to be unambiguous and reliable so that operator error or misinterpretation is avoided. The system would include the following control board indicators:

> An indication of upper region water level on each instrumented loop displaying water level in feet from 0 to -16 feet after compensation for any reactor coolant temperature and density effects. Indicator lights are included to indicate whether or not the pump in the loop is operating.

The Reactor Vessel Level Instrumentation is to be used in conjunction with a coolant subcooling readout to determine the state and transient behavior of the reactor coolant system. During normal operation, the reactor vessel level indicators would read off scale since the dynamic pressure drop due to coolant flow would be greater than the meter range. With all pumps shut down, the indicators will provide a direct indication of water level in the reactor vessel.

TVA will extend the range of incore thermocouples to give readout of fuel temperatures that could be expected if the core was partially uncovered.

1296 221

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

CLARIFICATION ITEMS

1. See the above response.

2. See the above response.

3. TVA's commitment is as submitted in response to the September 13, 1979, letter.

4. See the above response.

5. Existing instrumentation and subcooling monitor is used to give advance warning of the approach of inadequate core cooling.

6. See the above response.

CONTAINMENT ISOLATION (2.1.4)

POSITION

1. 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.

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

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

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

CLARIFICATION

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential systems should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation

1296 223

CONTAINMENT ISOLATION (2.1.4)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

The Sequoyah Nuclear Plant Meets all of the NRC positions concerning containment isolation. Specific information pertaining to each of the psoitions is given below.

RESPONSE

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

2. TVA has undertaken 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, (d) modify design if required, and (e) report results to NRC.

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

3. The Sequoyah Nuclear Plant design complies with NRC requirements on the automatic isolation of nonessential systems.

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

1296 224

CONTAINMENT ISOLATION (2.1.4)

CLARIFICATION ITEMS

1. Qualified diverse containment isolation signals are provided at Sequoyah.

2. As specified in the above response, an evaluation of essential and non-essential systems has be performed and Sequoyah complies with NRC requirements. This information will be made available for NRC review.

3. See section 3 of the above response.

4. See section 4 of the above response.

1296 225

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

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 the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

CLARIFICATION

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

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

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah does not use external recombiners or purge systems for post-accident combustible gas control.

RESPONSE

This requirement is not applicable to Sequoyah.

The Sequoyah design has a manually actuated ESF recombiner system inside containment which is redundant and fully qualified (see FSAR Section 6.2.5).

CLARIFICATION ITEMS

1. Not applicable to Sequoyah Nuclear Plant.
2. Not applicable to Sequoyah Nuclear Plant.
3. Not applicable to Sequoyah Nuclear Plant.
4. See the above response.
5. Not applicable to Sequoyah Nuclear Plant.

CAPABILITY TO INSTALL HYDROGEN RECOMBINER

AT EACH LIGHT WATER NUCLEAR POWER PLANT (2.1.5.c)

POSITION

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 sheilding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

CLARIFICATION

1. This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.B, "Design review for Plant Shielding."

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiner use. Action taken on this requirement should be submitted by January 1, 1980.

1296 228

CAPABILITY TO INSTALL HYDROGEN RECOMBINER

AT EACH LIGHT WATER NUCLEAR POWER PLANT (2.1.5.c)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

This requirement is not applicable to Sequoyah

RESPONSE

The Sequoyah design has an ESF recombiner system inside containment which is redundant and fully qualified (see FSAR Section 6.2.5) and is manually actuated from the main control room.

CLARIFICATION ITEMS

1. Combustible gas control has been accounted for in the Sequoyah design as stated in the above response.
2. Their is no personnel exposure associated with recombiner use at Sequoyah.
3. Procedures for recombiner use are adequate for Sequoyah Nuclear Plant and no upgrading is required.

1296 229

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 the NRC.

2. Continuing Leak Reduction

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

Response

TVA will investigate practical leakage reduction measures on systems which may contain radioactive fluids post-LOCA and will examine such systems as the residual heat removal (normal letdown path), containment spray and safety injection (recirculation mode), chemical volume and control, sampling, and waste disposal systems.

This examination will include a study of valve stem packing leakoffs, rotating seals on equipment, gasketed connections or joints, drain pipes to open connections, and building drainage systems.

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

1296 230

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATI.JN OF EQUIPMENT FOR SPACES/SYSEMS WHICH MAY B: USED IN POST ACCIDENT OPERATIONS (2.1.6.B)

POSITION

With the assumption of a post-accident release of 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 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, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

CLARIFICATION

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. In order to assure that personnel can perform necessary post-accident operations in the vital areas we are providing the following guidance 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 Plant 15.6.5. with appropriate decay times based on plant design.

1296 231

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 injected by HPCI and LPCI.

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 gas 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 gas space in the primary coolant system.

2. Dose Rate Criteria

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time 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

1296 232

with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. 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: $\leq$15mr/hr. 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 a regular basis, but 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 where occupancy may be needed often but not continuously.

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL

QUALIFICATION OF EQUIPMENT FOR SPACES/SYSEMS WHICH

MAY BE USED IN POST ACCIDENT OPERATIONS (2.1.6.B)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah Nuclear Plant is designed to mitigate major design basis events with no access outside the main control room required. Although the plant was not designed for access outside the control room, the current design may allow considerable capability for access for short times. TVA is evaluating the shielding requirements and will make design changes in shielding if the evaluation identifies feasible modifications which would significantly enhance desirable access.

RESPONSE

The Sequoyah 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 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 auxiliary 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 has 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 in the reactor containment.

1296 234

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS (2.1.6.b)

RESPONSE (cont.)

TVA will calculate the source terms for the sump water recirculating piping, pumps, and valves installed in the auxiliary building. TVA will then identify the vital areas in the auxiliary 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 and gamma shine intensities will then be calculated for the vital auxiliary building areas.

CLARIFICATION ITEMS

1. As specified in the above response, the Sequoyah design bases include the assumption of TID14844 sources.

Sequoyah Nuclear Plant will meet the requirements of GDC 19.

1296 235

AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM (AFWS) (2.1.7.A)

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 a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

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

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

1296 236

CLARIFICATION

Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.

2. Automatic initiation signals and circuits shall satisfy the single failure criterion.

3. Testability of the initiating signals and circuits is requied.

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

5. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not comprimise the emergency diesel generating capacity.

6. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.

7. Other Considerations

   a. For those designs where instrument air is needed for operation, appropriate electric power must also be supplied via the emergency power sources.

1296 237

AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM (AFWS) (2.1.7.A)

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 a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

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

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

CLARIFICATION

Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.
2. Automatic initiation signals and circuits shall satisfy the single failure criterion.
3. Testability of the initiating signals and circuits is requied.
4. Initiating signals and circuits shall be powered from the emergency buses.
5. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not comprimise the emergency diesel generating capacity.
6. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.
7. Other Considerations
   a. For those designs where instrument air is needed for operation, appropriate electric power must also be supplied via the emergency power sources.

1296 239

AUTO INITIATION OF AUXILIARY FEEDWATER (AFW)
2.1.7.A

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah complies with all of the requirements of 2.1.7.A.

Response

The auxiliary feedwater system is automatically initiated by redundant, coincident logic to preclude loss of function due to a single failure and to provide on line testability. The auxiliary feedwater system and initiating logic are described in TVA's response to NRC-OIE Bulletin 74-06A and in Sequoyah FSAR Section 10.4.7.2. The auxiliary feedwater control circuitry including the automatic initiating circuitry is safety-grade, Class 1E, and is powered from a power source connected to the emergency power system. Each auxiliary feedwater pump has manual initiation capability independent of the automatic initiation. The ac motor-driven pumps and valves are included in the automatic alignment of the loads to the emergency power system.

CLARIFICATION ITEMS

1. Automatic and manual initiation of AFW are provided at Sequoyah.
2. The AFW system is automatically initiated by redundant, coincident logic to preclude loss of function due to a single failure.
3. On line testability is provided.
4. Initiating signals are powered from the emergency power system.
5. The ac motor driven pumps and valves are included in the automatic alignment of loads to the emergency power system.
6. Manual initiation capability is provided independent of the automatic initiation.
7. Appropr electric power is supplied via the emergency power system for all valves where control air is needed for operation.

AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS (2.1.7.b)

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.

CLARIFICATION

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. Safety-Grade (Long-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. Other

1. For the Short-Term the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As

1296 241

a fall-back position, one auxiliary feed water flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feed water channel should provide an indication of feed flow with an accuracy on the order of ± 10%.

1296 242

AUXILIARY FEEDWATER FLOW INDICATION
2.1.7.B

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

AFW flow indication at Sequoyah is not calssified as safety grade; however, the components and design are similar to those used for safety grade systems. The AFW flow instrumentation channels are powered from the emergency buses.

Response

Auxiliary feedwater flow is indicated in the main control room for each of the four steam generators. The flow indication has not been classed as safety grade; however, it utilizes the same type of transmitters which are used in other safety grade circuits. The transmitters are mounted on two separate seismically qualified panels and powered from power sources connected to the emergency power system. The cables are in low level signal trays and are kept separate from all power cables. In addition, the total flow from the turbine driven auxiliary feedwater pump is indicated in the main control room. The auxiliary feedwater flow instrument channels are powered from the emergency buses consistent with the diversity requirements of the auxiliary feedwater system.

CLARIFICATION ITEMS

A. Control Grade (Short Term)

1. AFW flow indication to each steam generator does not satisfy the single failure criterion. (See item C.1)

2. Testability of the AFW flow indication channels is provided. (See item C.1)

3. The AFW flow instrument channels are powered from the emergency buses.

B. Safety Grade (Long Term)

1. AFW flow indication at Sequoyah is not classified as safety grade; however, the components and design are similar to those used for safety grade systems. The AFW flow instrumentation channels are powered from the emergency buses.

C. Other

1. The AFW flow indication channels do not by themselves satisfy the safety-grade requirements; however, the components and design are similar to those used for safety grade systems. In addition, the steam generator water level indication is safety-grade and satisfies the single failure and testability requirements. The steam generator water level provides backup indication of feedwater flow.

2. Each AFW flow instrument channel provides an indicaiton of feed flow with an accuracy on the order of ±10 percent.

IMPROVED POST-ACCIDENT SAMPLING CAPABILITY (2.1.8.a)

POSITION

A design and operational review of the react.. 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 quantify (less than 2 hours) 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 auxilia... building and possible contamination and direct radiation from airborne ...uents. If the review indicates that the analyses required cannot be ...erformed 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.

DISCUSSION

The primary purpose of implementing Improved Post-Accident Sampling Capability is to improve efforts to assess and control the course of an accident by:

1. Providing information related to the extent of core damage that has occurred or may be occurring during an accident;

2. Determining the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment;

1296 244

3. Providing information on coolant chemistry (e.g., dissolved gas, boron and pH) and containment hydrogen.

The above information requires a capability to perform the following analyses:

1. Radiological and chemical analyses of pressurized and unpressurized reactor coolant liquid samples;

2. Radiological and hydrogen analyses of containment atmosphere (air) samples.

CLARIFICATION

A. The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

B. The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),

2. hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,

3. dissolved gases (i.e., H_2 , O_2) and boron concentration of liquids.

4. pH of liquids,

or have in-line monitoring capabilities to perform the above analysis. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

C. During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provisions for purging samples lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose meterial in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

3. If changes or modifications to the existing sampling system are required, the seismic design and quality group classification or sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and piping downstream of the second isolation valve can be designed to quality Group D and nonseismic Category I requirements.

D. The licensee"s radiological sample analysis capability should include provisions to:

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 uCi/gm to the upper levels indicated here.

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

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels.

E. The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

F. In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonable achievable and not in excess of DGC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radiological and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling etc. shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

1296 246

4. High range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

G. The licensee shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section.

1296 247

IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

2.1.8.A

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

A design and operational review of the reactor coolant and containment atmosphere sampling systems and analysis facilities is being performed.

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.

To enhance the capability at Sequoyah for post-LOCA sampling TVA will:

1. Make provisions for sampling water from the reactor coolant system and the residual heat removal system for the degraded accident condition.

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

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

CLARIFICATION ITEMS

A. TVA will provide the capability to obtain (within one hour) Reactor Coolant samples and containment air samples under accident conditions. This capability will be provided by January 1, 1981.

B. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples, and a design review and procedural modifications (if necessary) will be completed by January 1, 1980. TVA will provide, as practical, onsite radiological and chemical analysis capacities in order to quantify the following:

1. core damage (RES)

2. hydrogen level in containment

3. dissolved gases and boron content (RCS)

4. pH (RCS)

C. Provisions will be made:

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

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

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

D. The radiological sample analysis capability will include provisions to:

a. Identify and quantify isotopes to levels corresponding to the source terms given in item 2.1.6.B. The ability to dilute samples and to measure nuclide concentrations as low as 1 μCi/gm will be provided.

b. Restrict background levels in the health physics laboratory.

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

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

F. Procedural changes and plant modifications will be made to assure that radiation exposures are as low as reasonably achievable. TVA will ensure that these criteria are met using the criteria identified.

G. TVA will demonstrate the capability to obtain and analyze a sample containing radio isotopes according to the criteria discussed above.

1296 249

INCREASED RANGE OF RADIATION MONITORS (2.1.8.b)

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 normal condition (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.

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.

DISCUSSION

The January 1, 1980 requirement, were specifically added by the Commission and were not included in NUREG-0578. The purpose of the interim January 1, 1980 requirement is to assure that licensees have methods of quantifing radioactivity releases should the existing effluent instrumentation go offscale.

CLARIFICATION

1. Radiological Noble Gas Effluent Monitors

   A. January 1, 1980 Requirements

   Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points, (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/filter paper). Existing sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey

1296 251

instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (μCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seismically qualified equipment and equipment meeting IEEE-279 is not required.

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

1. Noble Gas Effluents

   a. System/Method description including:

      i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

      ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

      iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control room read-out is perferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

1296 252

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/ analysis including:

i) Procedures for minimizing occupational exposures

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

B. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i, 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

B. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.A above.

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.2. It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function.

1296 254

2. Radioiodine and Particulate Effluents

A. For January 1, 1980 the licensee should provide the following:

1. System/Method description including:

a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b) Monitoring/sampling location.

c) Method to be used for retrieval and handling of sampling media to minimize occupational exposure.

d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

2. Procedures for conducting all aspects of the measurement analysis including:

a) Minimizing occupational exposure

b) Calculational methods for determining release rates

c) Procedures for dissemination of information

d) Calibration frequency and technique

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL

ACCIDENTAL RADIOACTIVITY RELEASES

- Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

- Examples of major elements of a highly radioactive effluent release special procedures (noble gas).
    - Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.
    - Provide shielding to minimize background interference.
    - Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.
    - Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

1296 256

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

- NOBLE GASES ONLY
- RANGE: (Overlap with Normal Effluent Instrument Range)
  - UNDILUTED CONTAINMENT EXHAUST 10^{+5} μCi/CC
  - DILUTED (>10: 1) CONTAINMENT EXHAUST 10^{+4} μCi/CC
  - MARK I BWR REACTOR BUILDING EXHAUST 10^{+4} μCi/CC
  - PWR SECONDARY CONTAINMENT EXHAUST 10^{+4} μCi/CC
  - BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES 10^{+3} μCi/CC
  - OTHER BUILDINGS (E.G., RADWASTE) 10^{+2} μCi/CC
- NOT REDUNDANT - 1 PER NORMAL RELEASE POINT
- SEISMIC - NO
- POWER - VITAL INSTRUMENT BUS
- SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979
- DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)
- QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

1296 257

TABLE 2.1.8.b.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R. G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R.G. 1.97 REV. 2 AND ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 KEV photon Xe-133 is the major noble gas present.

1296 258

INCREASED RANGE OF RADIATION MONITORS
2.1.8.B

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah will comply with the requirements of 2.1.8.B by January 1, 1981, except that high level radiation monitors will be located outside the annulus instead of inside containment. Interim measures will be provided before fuel loading in the respective units for quantifying high level releases.

Response

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

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

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

Interim Procedures for Quantifying High Level Accidental Radioactivity Releases

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

CLARIFICATION ITEMS

1. Noble Gas Effluent Monitors

   A. Requirements for January 1, 1980 - TVA will provide an instrument to monitor gross releases of radioactivity from the shield building vent. Our present shield building vent monitor provides a gaseous sample for laboratory analysis. Special procedures will be developed for estimating noble gas effluent in the event present instrumentation saturates. A description of these systems and methods will be made available to the NRC.

1296 259

B. By January 1, 1981, TVA will provide high range noble gas effluent monitors for all identified release paths. This monitor will meet the requirements of Table 2.1.8.B.2. Information requested on these monitors will be made available to the NRC.

2. Radioiodine and Particulate Effluents

A. Requirements for January 1, 1980 - Methods used to sample radioiodine and particulates will be provided.

B. By January 1, 1981, TVA will provide the capability to continuously sample effluents and onsite analysis for radioiodine and particulates with state-of-the-art equipment. The requested information will be made available to the NRC.

3. Containment Radiation Monitors

By January 1, 1981, TVA will provide two radiation monitors outside the annulus which meet the intent of the requirements of Table 2.1.8.B.3.

1296 260

IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (2.1.8.c)

POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

CLARIFICATION

Use of Portable versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

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

1296 261

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

Iodine Filters and Measurement Techniques

A. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too must activity collected on the sampling cartridge.

B. By January 1, 1981:

The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble bases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

IMPROVED IN-PLANT IODINE INSTRUME TATION UNDER ACCIDENT CONDITIONS
2.1 8.C

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah ahs low-volume portable air monitore equipped with a charcoal filter to absorb iodine isotopes. These filters will be analyzed in the health physics laboratory. This capability meets the requirements of 2.1.8.C.

Response

Sequoyah has portable low-volume air samplers, each equipped with a particulate filter followed by a charcoal absorber to collect iodine isotopes. The particulate filter will be counted in the health physics laboratory for gross activity and the charcoal absorber 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 in-plant 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 in-plant air spaces incorporated in the Health Ohysics Laboratory Instruction Manual and the Radiation Control Instruction Manual.

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

CLARIFICATION ITEMS

A. Requirements before fuel loading.

Sequoyah has portable low-volume air samplers equipped with a charcoal absorber to collect iodine. The filters are removed and taken to the health physics laboratory for analysis. Single channel analyzers are not provided for detection of iodine.

B. January 1, 1981, requirements.

Sequoyah's low-volume air samplers and health physics lab meet these requirements.

1296 263

TRANSIENT AND ACCIDENT ANALYSIS (2.1.9)

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 fome 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 natura 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 epeated taking credit for correct operator action. Th additic al cases will provide the basis for developing appropriate ei ergenc pr cedures. These calculations should also provide the analyt al basis for design of any additional instrumentation needed t provide ope ors with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 each FSAR. The analyses shall include a single active failure for ea n 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 a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, 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

1296 264

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 MASS 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 comparison, together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

DISCUSSION

The scope of the required transient and accident analysis is discussed in NUREG-0578. The schedule for these analyses is included in NUREG-0578 and is reproduced in the Implementation Schedule attachment to this letter. The Bulletins and Orders Task Force has been implementing these required analyses on that schedule. The analysis of the small break loss of coolant accident has been submitted by each of the owners groups. These analyses are presently under review by the B&O Task Force. The scope and schedule for the analysis of inadequate core cooling have been discussed and agreed upon in meetings between the owners groups and the B&O Task Force, and are documented in the minutes to those meetings.

The analysis of transients and accidents for the purpose of upgrading emergency procedures is due in early 1980 and the detailed scope and schedule of this analysis is the subject of continuing discussions between the owners groups and the B&O Task For

1296 265

TRANSIENT AND ACCIDENT ANALYSIS 2.1.9

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA is pursuing the required analyses and the development of new procedures and training guidelines with other utilities through the Westinghouse owners group. We doubt that the extremely ambitious implementation schedule of NUREG-0578 can be met without extraordinary effort on all parts.

Response

TVA is pursuing the required analyses and the development of new procedures and training guidelines with other utilities through the Westinghouse TMI Owners Group.

The transient and accident analyses should use realistic codes and include event tree analyses. The analyses should consider permutations and combinations of operator errors and equipment failures, including single failures in multiple systems and multiple operator errors. The operating procedures and operator training that will evolve from these analyses are essential to enhancing safety by improving reactor operator performance during transient and accident conditions.

Small break loss-of-coolant accident analyses have been performed and submitted to NRC in WCAP 9600. The report presents a comprehensive study of Westinghouse system response to small breaks. Westinghouse has already discussed continuing efforts aimed at improving emergency operating procedure guidelines with the NRC.

Inadequate core cooling is an item where further definition of the scope, such as system failure and operator error assumptions, is needed from the NRC. At present model preparation is in progress to permit response to identified action. Westinghouse does plan to perform pretest calculations of the LOFT tests when we are provided with the necessary input information.

The purpose of this action is to improve the performance of reactor operators during transient and accident conditions. The primary concern is that the operator training and emergency operating procedures are based on the conservative plant FSAR Chapter 15 analyses. Chapter 15 should continue to be used for design basis analyses since these show the most limiting initial approach to safety limits. What is needed is to evaluate the longterm consequences of accidents using realistic assumptions incorporating the effects of the following:

1. Operator's failure to act when required.
2. Operator's inappropriate actions during an accident.
3. Additional failures.
4. Selected system operations (e.g., restarting of RCP's etc.)

1296 266

Appropriate changes can then be incorporated into the existing procedures, designs, and training programs.

Development of the models to incorporate such effects is in itself a longterm effort before detailed analyses can be run. Significant interaction between industry and the NRC is required to agree on the assumptions, bases, appropriate actions or misactions to be modeled, and best estimate boundary conditions.

Based on TVA's perception of NRC intent, the proposed implementation schedule in NUREG 0578 is extremely ambitious. We believe that it cannot be met without an extraordinary effort on the part of NSSS vendors, utilities, and the NRC staff. While we agree with the urgency attached to this effort, we caution that undue haste, just to meet the implementation schedule, is unwarranted.

CONTAINMENT PRESSURE INDICATION

POSITION

A continuous indication of containment pressure should 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.

CLARIFICATION

1. The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.
2. The containment pressure monitor shall be installed by January 1, 1981.

1296 268

CONTAINMENT PRESSURE INDICATION 2.1.9(a)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah will comply with all of the requirements of this position before January 1, 1981.

Response

Four qualified, continuous indications of the containment pressure are provided in the main control room. The 5 psig negative pressure requirement is not applicable to Sequoyah since qualified vacuum relief of the containment maintains the pressure at greater than negative 0.5 psig. The negative range of the existing pressure indicators envelopes this negative 0.5 psig limit. Redundant, continuous containment pressure indication with a range up to four times the design pressure of the steel containment will be provided.

CLARIFICATION ITEMS

1. The monitors will meet the applicable design requirements for qualification, redundancy and testability.

2. The monitors will be installed and operational by January 1, 1981.

1296 269

CONTAINMENT HYDROGEN INDICATION

POSITION

A continuous indicaton 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.

CLARIFICATION

1. The containment hydrogen indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy and testability.

2. The containment hydrogen indication shall be installed by January 1, 1981.

1296 270

CONTAINMENT HYDROGEN INDICATION 2.1.9(b)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

Sequoyah has redundant safety-grade hydrogen analyzers located in the annulus. These monitors have a range of 0 to 10 percent hydrogen concentration. Sequoyah complies with all of the requirements of this NRC position.

Response

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

CLARIFICATION ITEMS

1. The hydrogen analyzers of Sequoyah meet the applicable requirements for qualification, redundancy and testability.

2. These analyzers are installed and operational.

CONTAINMENT WATER LEVEL INDICATION

POSITION

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

CLARIFICATION

1. The narrow range sump level instrument shall monitor the normal containment sump level vice the containment emergency sump level.

2. The wide range containment water level instruments shall meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following a Accident).

3. The narrow range containment water level instruments shall meet the requirements of Regulatory Guide 1.89 (Qualification of Class IE Equipment of Nuclear Power Plants).

4. The equivalent capacity of the wide range PWR level instrument has been changed from 500,000 gallons to 600,000 gallons to ensure consistency with the proposed revision to Regulatory Guide 1.97.

5. The containment water level indication shall be installed by January 1, 1981.

1296 272

CONTAINMENT WATER LEVEL INDICATION 2.1.9(c)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

The sump water level is indicated by four separate qualified, and continuous level instruments with readout in the main control room. These instruments provide adequate indication of the water level in the sump. Sequoyah complies with all of the requirements of this NRC position.

Response

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

CLARIFICATION ITEMS

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

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

3. The narrow range sump level instrument meet the applicable requirements for qualification, redundancy, and testability.

4. If 600,000 gallons of water were introduced into containment, in addition to the entire fluid volume of the reactor coolant system, safety injection accumulators, and a total ice melt, the containment water level would not exceed the design basis for the wide range water level monitor.

5. The sump water level monitors are installed and operational.

1296 273

REACTOR COOLANT SYSTEM VENTING

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 deomonstrating 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 or terminating vent usage shall be discussed.

CLARIFICATION

A. General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10CFR50.46 (LOCA criteria) and 10CFR50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

1296 274

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures 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 based on the following criteria: (1) assurance that the plant can meet the requirements of 10CFR50.46 and 10CFR50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the BWR owners group has suggested that the present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be demonstrated. In addition the ability of these systems to meet the same requirements as the PWR vent systems must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The locations for PWR Vents are as follows:

   a. Each PWR licensee should provide the capability to vent the reactor vessel head.

   b. The reactor vessel head vent should be capable of venting non-condensible 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). Additional venting capability is required for those portions of each hot leg which can not be vented through the the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such a procedure is required by January 1981.

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

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time i.e., a vent capable of venting a gas volume of 1/2 the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

1296 276

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

REACTOR COOLANT SYSTEM VENTING 2.1.9(d)

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA will provide the capability to vent the reactor vessel head by January 1, 1981. The design for this vent will be made available for NRC review by January 1, 1980.

Response

TVA will provide the capability to vent the reactor vessel head in addition to the existing venting capability from the pressurizer. The new reactor vessel head vent system will meet all of the NRC requirements.

It is, of course, not feasible to directly vent the reactor coolant system high points in the U-tubes of the steam generators. This venting capability is not required.

CLARIFICATION ITEMS

A. Procedures for use of the reactor vessel head vent at Sequoyah will be made available to the NRC before January 1, 1981.

B. (Not applicable to Sequoyah)

C. PWR Vent Design Consideration

1. a) A reactor vessel head vent will be installed by January 1, 1981, to provide the capability to vent noncondensible gas from the reactor coolant system.

   b) No additional vents are required. Natural circulation in the primary system will ensure that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed.

   c) Venting of the pressurizer is provided as part of the Sequoyah design.

2. Appropriate design considerations will be implemented in design of the reactor vessel head vent.

1296 278

SHIFT SUPERVISOR RESPONSIBILITIES (2.2.1.a)

POSITION

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

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

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

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

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

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

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

CLARIFICATION

The attachment provides clarification to the above position.

1296 279

Attachment

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.A)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V. P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V. P. for Operations.

1296 280

SHIFT SUPERVISOR RESPONSIBILITIES (2.2.1.a)

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The requirements are to be implemented by OL for SNP unit 1. The duties of the shift supervisor, as discussed in NUREG-0578, are performed by the assistant shift engineer on each unit. The V.P. for Operations is the Manager of Power Operations. SRO training is specified in the SNP FSAR 13.2 which meets the intent of section 5.2.1.8.

RESPONSE

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

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

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 responses, we will use the term assistant shift engineer for shift supervisor.

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

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

1296 281

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

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

4. The administrative duties of the shift supervisor 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. The following actions have already been taken:

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

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

CLARIFICATION

Not required.

1296 282

SHIFT TECHNICAL ADVISOR (Section 2.2.1.b)

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 a 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 assuring safe operation of the plant, including the review and evaluation of operating experience.

DISCUSSION

The NRC Lessons Learned Task Force has recommended the use of Shift Technical Adviors (STA) as a method of immediately improving the plant operating staff's capabilities for response to off-normal conditions and for evaluating operating experience.

In defining the characteristics of the STA, we have used the two essential functions to be provided by the STA. These are accident assessment and operating experience assessment.

1. Accident Assessment

   The STA serving the accident assessment function must be dedicated to concern for the safety of the plant. The STA's duties will be to diagnose off-normal events and advise the shift supervisor. The duties of the STA should not include the manipulatin of controls or supervision of operators. The STA must be available, in the control room, within 10 minutes of being summoned.

   The qualifications of the STA should include college level education in engineering and science subjects as well as training in reactor operations both normal and off-normal. Details regarding these qualifications are provided in paragraphs A.1, 2 and 3 of Enclosure 2 to our September 13, 1979 letter. In addition, the STA serving the accident assessment function must be cognizant of the evaluations performed as part of the operating experience assessment function.

2. Operating Experience Assessment

The persons serving the opeating experience assessment function must be dedicated to concern for the safety of the plant. Their function will be to evaluate plant operations from a safety point of view and should include such assignments as listed on pages A-50 and A-51 of NUREG-0578. Their qualifications are identical to those described previously under accident assessment and collectively this group should provide competence in all technical areas important to safety. It is desirable that this function be performed by onsite personnel.

CLARIFICATION

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgements by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant.

1296 285

SHIFT TECHNICAL ADVISOR (2.2.1.b)

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The shift technical advisor requirements are to be implemented by January 1, 1980, or by initial criticality. The shift technical advisor training will not be complete until January 1981; however, minimum training requirements will be completed.

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, license 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.

CLARIFICATION

1. In addition to the accident assessment function, the shift technical advisor will be cognizant of information determined by the TVA Operating Experience Review Group.

2. The shift technical advisor will be independent of duties that detract from his primary functions or dilute his dedication to these primary functions. The shift technical advisor will be an addition to the previously defined operating staff.

3. Although the shift technical advisor will not be completely trained for his duties by January 1, 1980, the STA will be a full-time shift employee who will be available within 10 minutes of being summoned during any shift.

4. The shift technical advisor will be on duty by January 1, 1980, and training requirements will be met by January 1, 1981. The shift technical advisors on duty by January 1, 1980, will provide additional accident and operating experience assessment.

SHIFT AND RELIEF TURNOVER PROCEDURES (2.2.1.c)

POSITION

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

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

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

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

   (what to check and criteria for acceptable status shall be included on the checklist);

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

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or 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 verification of system alignments).

CLARIFICATION

No clarification provided.

SHIFT AND RELIEF TURNOVER PROCEDURES (2.2.1.c)

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The new shift and relief turnover procedures have been developed and will be implemented for fuel load.

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 and signed 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. 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 could degrade a safety system. In addition, a system will be established to evaluate the effectiveness of the turnover procedures.

1296 289

CONTROL ROOM ACCESS (2.2.2.a)

POSITION

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

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

2. Develop and implement procedures that establish a clear line of 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.

CLARIFICATION

No clarification provided.

ONSITE TECHNICAL SUPPORT CENTER (TSC) 2.2.2.b

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. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

CLARIFICATION

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

    A. Establish a TSC and provide a complete description,

    B. Provide plans and procedures for engineering/management support and staffing of the TSC,

    C. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC,

    D. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminmants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation to the control room),

    E. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

1296 291

CONTROL ROOM ACCESS (2.2.2.a)

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The new procedures have been developed and will be implemented for fuel load.

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.

F. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

G. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size & Staffing

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4 Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

1296 293

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be give to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum,

1296 294

data (historical in addition to current status) should be available to permit the assessment of:

Plant Safety Systems Parameters for:

- Reactor Coolant System
- Secondary System (PWRs)
- ECCS Systems
- Feedwater & Makeup Systems
- Containment

In-Plant Radiological Parameters for:

- Reactor Coolant System
- Containment
- Effluent Treatment
- Release Paths

Offsite Radiological

- Meteorology
- Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

1296 295

9. <u>Structural Integrity</u>

A. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

B. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. <u>Habitability</u>

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

A. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

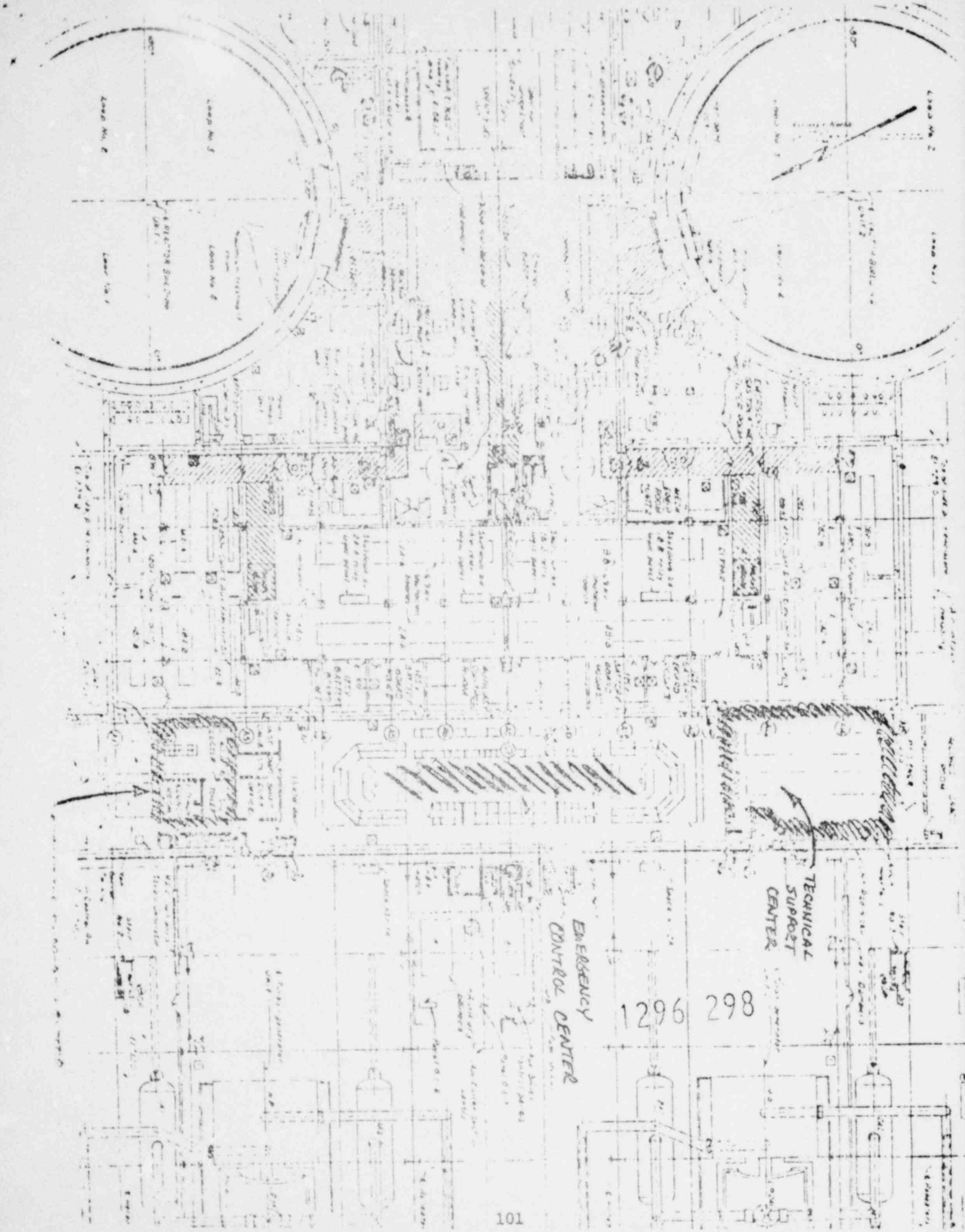
B. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

1296 296

C. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

D. Dose reduction measures such as breathing apparatus and potassium iodide tablets can not be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

1296 297



1296 298

ONSITE TECHNICAL SUPPORT CENTER (TSC) (2.2.2.b)

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The onstie technical support center (TSC) has been established and meets the criteria for the interim TSC, as well as many of the criteria for the permanent TSC.

RESPONSE

The onsite technical support center will be established on the same floor as the main control room (MCR) but outside of the MCR. It will be habitable to the same extent as the MCR and will have ready access to a complete set of as-built drawings. Reliable communications will be provided to the MCR The technical support center will be established before receipt of the operating license.

Prior to January 1, 1981, equipment will be installed in the support center to improve that plant monitoring capability of technical support personnel.

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

CLARIFICATION

1.A. The TSC description will be provided by January 1, 1980.

1.B. The plans and procedures for engineering/management support and staffing of the TSC will be provided by December 1, 1979.

1.C. The requested communications between the TSC and the control room, the emergency operations center in Chattanooga, and the NRC have been installed.

1.D. Portable radiation monitors will be provided for the TSC until permanent monitors are available.

1.E. This item will be addressed in our submittal of the TSC description.

1.F. Procedures will be provided for performing the accident assessment function from the control room should the TSC become uninhabitable and will be revised as the TSC is upgraded.

1.G. The long-range plan for upgrading the TSC will be submitted before January 1, 1981.

2. The TSC is located next to the control room (see attached sketch).

1296 299

3. The TSC can accommodate 25 persons. Specifics as to physical size and staffing will be provided (see item 1.A.).

4. The TSC activation is defined in the SNP Emergency Plan. The classification nomenclature of NUREG-0610 will implemented by July 1, 1980. The intent of the NUREG-0610 alert levels are implemented. Instrumentation in the TSC will provided in the submittals on TSC design (see Item 1.A.).

5. Design and capability of instrumentation for the permanent TSC will be provided before January 1, 1981.

6. Design criteria for TSC instrumentation will be provided before January 1, 1981.

7. The TSC data requirements will be addressed in a description of the upgraded TSC before January 1, 1981.

8. The TSC data transmission links to offsite centers is being discussed with the Westinghouse Owner's Group for TMI-II.

9.A. The TSC is located in a seismic Category I structure.

9.B. The backup plan for responding to an emergency from the control room will be defined (see item 1.F.).

10.A. The TSC is located in the control room complex. The permanent design meets GDC 19 and SRP 6.4 specifications, as well as TVA's radiological health and safety requirements.

10.B. Permanent monitoring systems will be provided. Procedures for response to high dose rates will be established by January 1, 1980.

10.C. Ventilation systems will be addressed in the TSA description submittals.

10.D. Dose reduction measures and breathing apparatus will be addressed in the TSA description submittals.

ONSITE OPERATIONAL SUPPORT CENTER (SECTION 2.2.2.c)

POSITION

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

CLARIFICATION

No clarification provided.

1296 301

ONSITE OPERATIONAL SUPPORT CENTER

SEQUOYAH NUCLEAR PLANT UNIT 1 RESPONSE

SUMMARY

The operational support center has been established and meets the criteria.

RESPONSE

The operational support center with communications to the main control room has been established. This center is located on the control room floor next to the shift engineers office (see figure in response to item 2.2.2.b.). The plant Radiological Emergency Plan will be revised by January 1, 1980, to reflect this center and to specify the personnel who will report in the event of an emergency.

1296 302