TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

January 11, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Rubenstein:

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

Enclosed are forty copies of revisions to TVA's revised response to NUREG 0578, Short Term Lessons Learned Requirements, for Sequoyah Nuclear Plant. These revisions reflect the current status of TVA's commitments on NUREG 0578. We expect to submit additional revisions before February 1, 1980, which revise the previous submittals of procedures and operating instructions.

For your convenience, the enclosed revised pages replace the corresponding pages in the revised response to NUREG 0578 submitted by my letter to you dated October 31, 1979. Additional revised pages were submitted by my letter to you dated November 21, 1979.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure

?cys Advanced
To M Williams

1759 109

B001
SE 1/10

An Equal Opportunity Employer

8001160543

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

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 is developing 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.

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force, submitted the "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," December 13, 1979. TVA considers this program to be responsive to the requirements for generic performance testing. The EPRI Program Plan provides for a completion of the essential portions of the test program by July, 1981. TVA will be participating in the EPRI program to provide program review and to supply plant specific data as required.

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.

JAN 11

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. 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 margin to saturation will be continuously displayed on a computer output trend recorder in the main control room.

TVA has received a procedure for detection of inadequate core cooling with currently available instrumentation from the Westinghouse owners group (of which TVA is a member). This procedure is being reviewed before incorporation as a plant procedure.

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. 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.

JAN 11 1980

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 have been 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 and between the top and bottom of the reactor vessel.

RESPONSE

Analysis and procedures for the detection of inadequate core cooling using existing instrumentation have been developed in conjunction with the Westinghouse Owners' Group. TVA is reviewing these procedures before incorporating them into plant procedures.

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 and between the top and bottom of the reactor vessel. Refer to figure 2.1.3.b-1. This instrumentation will be designed and qualified in accordance with safety grade, Class IE, requirements including redundancy and emergency power.

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

1758 112 JAN 11 1980

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

ADDITIONAL INSTRUMENTATION

RESPONSE (cont.)

The Reactor Vessel Level Instrumentation System has differential pressure measurement across the upper region of the reactor vessel. The system utilizes two differential pressure cells measuring the pressure drop from the 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.

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

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

To provide the required accuracy for water level measurement, temperature measurements of the reference legs are provided. These measurements together with the reactor coolant temperature measurements are used to compensate the differential pressure 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 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.

1759 113

JAN 11 1980

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.

Two indications of reactor vessel narrow range level displaying water level in feet from 0 to -40 feet after compensation for the effects of the reactor coolant temperature and density. Indicator lights are provided to show whether or not the pump in that loop is operating.

Two indications of reactor vessel wide range pressure drop displaying pressure drop in percent of the normal operating pressure drop. A switch is provided to change the meter range depending on the number of reactor coolant pumps running. Ranges will be precalibrated for different numbers of coolant pumps running. The calibration will be done during hot shutdown. The differential pressure of the transducer will be +15 to -30 psi, and the instrument ranges will be compensated for differences between hot shutdown temperature and operating temperatures.

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. This readout will indicate when thermocouple temperatures are offscale high.

1759 114

JAN 11 1980

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 been performed and Sequoyah complies with NRC requirements.

The containment isolation system at SQN is designed to prevent the release of radioactive material to the environment after an accident while ensuring that systems important for postaccident mitigation are operational. Table 2.1.4-1 shows the different isolation signals and the parameters that initiate each signal.

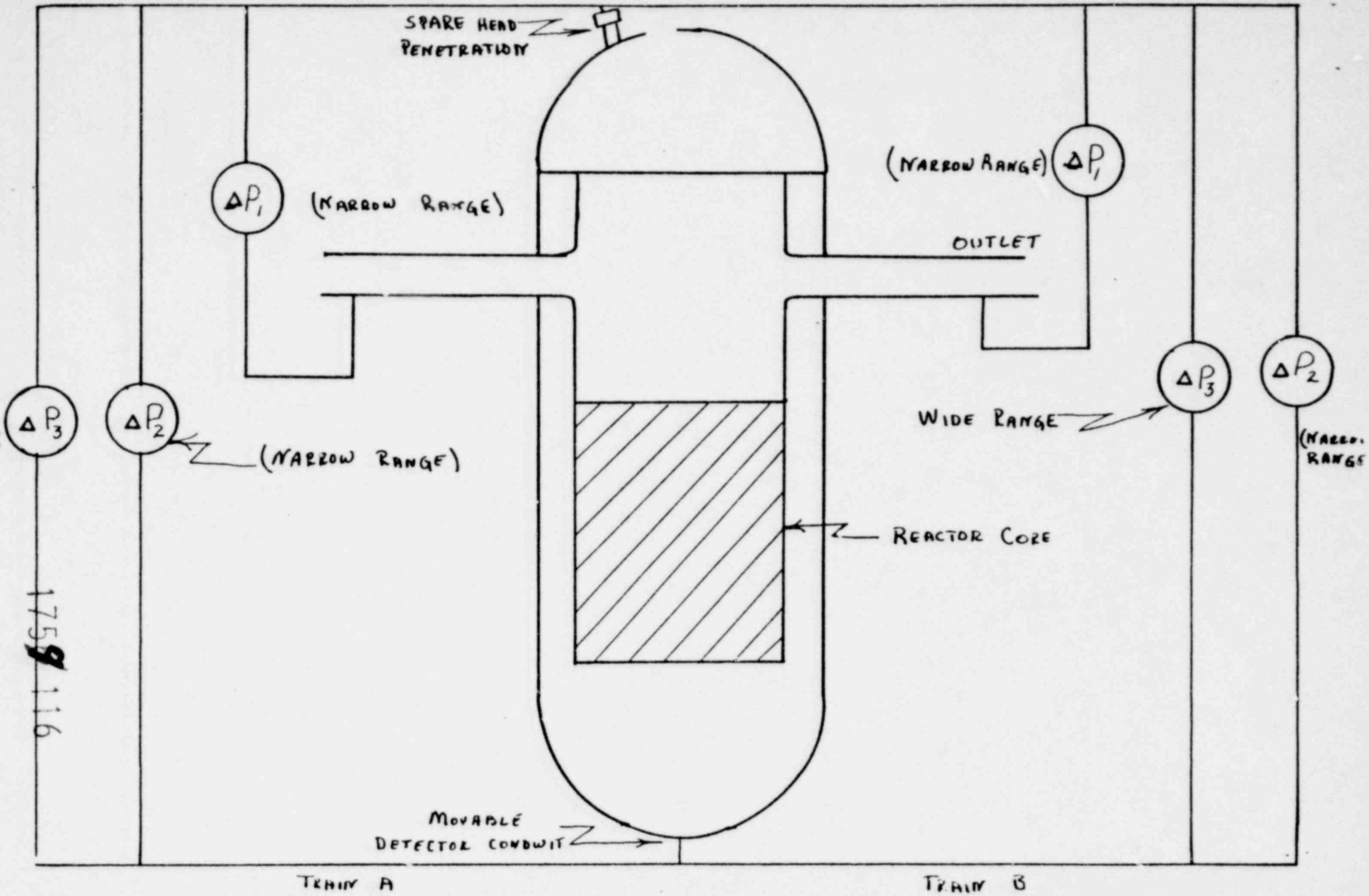
Isolation at SQN is provided on the following three levels:

1. Nonessential systems - These systems are not required for postaccident mitigation and are isolated automatically upon receipt of a Phase A isolation signal.
2. Essential systems - This group consists of the emergency core cooling systems, the containment spray system, and postaccident H_2 monitors. These systems are not automatically isolated in the event of an accident. Remote manual valves are provided to permit isolation of these lines from the main control room if necessary.
3. Desirable systems - Systems that, while not required, significantly increase the plant's ability to cope with a small steam line break or LOCA. The systems are isolated automatically upon the receipt of a Phase B isolation signal (Table 1). The systems falling into this category are emergency raw cooling water to the reactor coolant pumps (RCP) and containment coolers, component cooling water to the RCP's and control air.

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

The containment isolation system is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. The initiating signal must be reset and each automatic valve individually opened by the operator. Resetting of an initiating signal will not cause a containment isolation valve to change position.

The isolation of ventilation lines and lines that carry potentially radioactive fluid outside containment during power operation received special consideration at SQN. The ventilation lines receive high radiation signals in addition to the Phase A or B isolation signals (Table 2.1.4-1). At present, the isolation of fluid lines that carry potentially radioactive material outside containment occurs upon the receipt of a Phase A signal. This isolation signal would preclude the type of releases of radioactive material that occurred at TMI. However, to provide an additional margin



REACTOR VESSEL WATER LEVEL MEASUREMENT SYSTEM

FIGURE 2.1.3.b-1

1759 116

of safety as identified in the TVA Nuclear Program Review as a result of TMI, TVA is adding radiation monitors that will automatically isolate each of these lines in the event of high radiation in the line. These monitors and associated isolation logic changes will be in the plant by May 1981.

3. See section 3 of the above response.

4. See section 4 of the above response.

JAN 11 1980

1754 117

TABLE 2.1.4-1

Isolation Signals

Phase A

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

SIS Initiation

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

Phase B Initiation

Manually - 2 of 4 hand switches or,
Automatically - 2 of 4 high-high containment pressure

Containment Ventilation Initiation

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

1759 118

JAN 11 1980

SYSTEMS INTEGRITY FOR HIGH RADIOACTIVITY

SECTION 2.1.6.a

SEQUOYAH NUCLEAR PLANT RESPONSE

SUMMARY

TVA has investigated practical leakage reduction measures on systems which may contain radioactive fluids. Procedures for reducing and quantifying leakage from liquid systems are presently in the review and approval cycle. These procedures will be submitted for NRC review.

RESPONSE

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

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

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

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

Procedures for reducing and quantifying leakage from liquid systems are presently in the review and approval cycle. These procedures will be submitted to NRC. These procedures were written in compliance with the guidelines listed below.

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

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

3. Inspection will be performed quarterly except where accessibility is limited by radiation exposure. These systems or portions of a system will be inspected during hot shutdown or other periods of accessibility.

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

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

6. Leakage for each quarterly test will be reported on an annual basis to NRC.

The systems identified for leakage checks are listed below.

a. Safety Injection
b. Containment Spray
c. RHR
d. Equipment and Floor Drain Systems (Primary Containment)
e. *Chemical and Volume Control System
f. Sampling

*Not required quarterly due to radiation exposure

Procedures for reducing leakage from gaseous systems are based on the surveillance and inservice testing programs. Identification of gaseous leakage is accomplished in response to any alarm from area radiation dectectors. Leakages will require immediate attention. All gaseous leakages will be "tracked" and be controlled.

CLARIFICATION ITEMS

Refer to response above.

JAN 11

1759 120

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 allows considerable capability for access for short times. TVA has evaluated the shielding requirements and will make design changes in shielding where the evaluation identified 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 main control room (MCR) being required. With this goal in mind, the plants were not specifically designed for any access outside the MCR. To specifically design for guaranteed access at any time in most parts of the auxiliary building is not feasible. However, the current designs allow considerable capability for access for short times if the entry time into the area can be selectively chosen.

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

Sequoyah is designed to mitigate major accidents without access to the plant outside the MCR. Two areas outside the MCR were identified which would be helpful in responding to an accident situation. One area is a control panel in the shutdown board area at elevation 734.0. The panel is immediately outside the MCR. There are no contained sources in this area and direct gamma doses will not cause any concern for access. The other area is the normal plant sampling station in the auxiliary building at elevation 690.0. Dose rates in the sample room were evaluated for various times into the accident. A representative value at one hour into the accident is 900 mr/minute. Sampling procedures for accident situations in the interim period until a redesigned sampling facility can

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.)

be installed take into account these calculated values. If samples are ever needed in an accident, the procedure will also utilize actual dose rate measurements to evaluate accessibility and occupancy times.

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

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.

JAN 11 1980

1759 122

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 has been performed. Modifications are expected to be complete by January 1, 1981.

RESPONSE

A design and operational review of the reactor coolant sampling systems and analysis facilities has been performed. TVA expects to complete required modifications by January 1, 1981, provided that equipment procurement/installation conflicts are not encountered. These modifications will make providions for sampling water from the reactor coolant system for the degraded accident condition.

The TVA review of reactor coolant sampling systems has determined that current facilities are not adequate to sample, handle, and analyze highly radioactive water and gas samples and that modification of existing facilities is impractical. As a result, TVA is negotiating a contract with Oak Ridge National Laboratory (ORNL) to develop and design such a facility. ORNL will establish the basic concepts, work with TVA to locate the facility in the plant, and prepare design criteria diagrams and physical arrangement drawings by June 1, 1980. ORNL will also prepare specification inputs and provide engineering support in the latter phases of implementation.

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. Refer to pages 52a through 52m for current sampling procedures.

ADDENDUM TO ITEM 2.1.8.a

RADIATION LEVELS FOLLOWING LOW POWER PHYSICS TESTING PROGRAM

NRC Concern

Evaluate the radiation levels that will exist after the low-power testing is completed (including that from crud deposits) to ensure that any requirements for physical alterations dictated by the Lessons Learned Task Force, Kemeny Commission, Rogovin Commission, or Task Action Plan can be implemented. State that the radiation exposure from these tests will not preclude any changes or additions or deletions from the plant.

TVA Response

TVA does not foresee that the radiation levels created by the low-power testing will prevent implementation of any requirements for physical alterations dictated by the Lessons Learned Task Force, Kemeny Commission, Rogovin Commission, or Task Action Plan as presently understood. The radiation exposure from these tests will not preclude any currently identified changes, additions, or deletions from the plant.

JAN 11 1980

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. 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.

CLARIFICATION ITEMS (cont'd.)

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

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 - A design study to assist in developing interim procedures for monitoring radioiodine and particulate effluents is underway. The procedures will be provided to the NRC when they are developed.

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.

JAN 11 1980

1759 126

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. Westinghouse has submitted a supplement to WCAP 9600, WCAP 9639, which addresses UHI plants to the NRC. TVA is currently reviewing this analysis and expects to complete the review in March 1980.

Westinghouse has submitted an analysis of inadequate core cooling to the NRC. TVA is currently in the process of reviewing this analysis and expects to complete the review in March 1980.

Westinghouse has submitted to the NRC an analysis of the LOFT tests. TVA is reviewing this analysis.

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 long-term consequences of accidents using realistic assumptions incorporating the effects of the following:

Response (cont'd.)

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

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

Westinghouse and the Division of Engineering Design will review Sequoyah's Emergency and Abnormal Operating Instructions by January 18, 1980.

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.

ADDENDUM TO ITEM 2.1.9

REVIEW OF OPERATING PROCEDURES

Figures A-2.1.9-1 and A-2.1.9-2 describe the way TVA has traditionally developed and reviewed operating procedures.

1759 129

JAN 11 1980

OPERATIONAL PROCEDURES DEVELOPMENT AND REVIEW
SEQUOYAH NUCLEAR PLANT

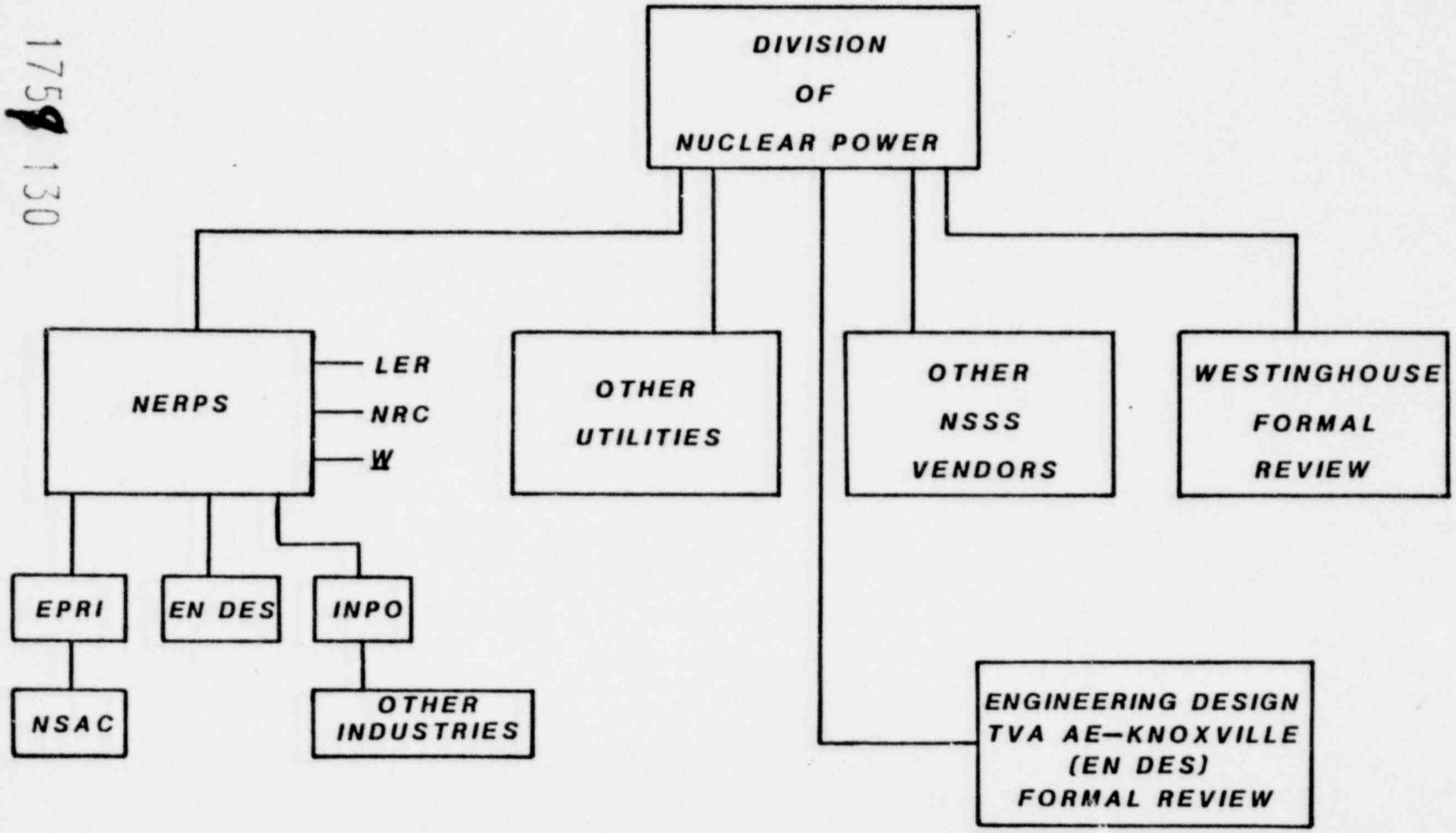
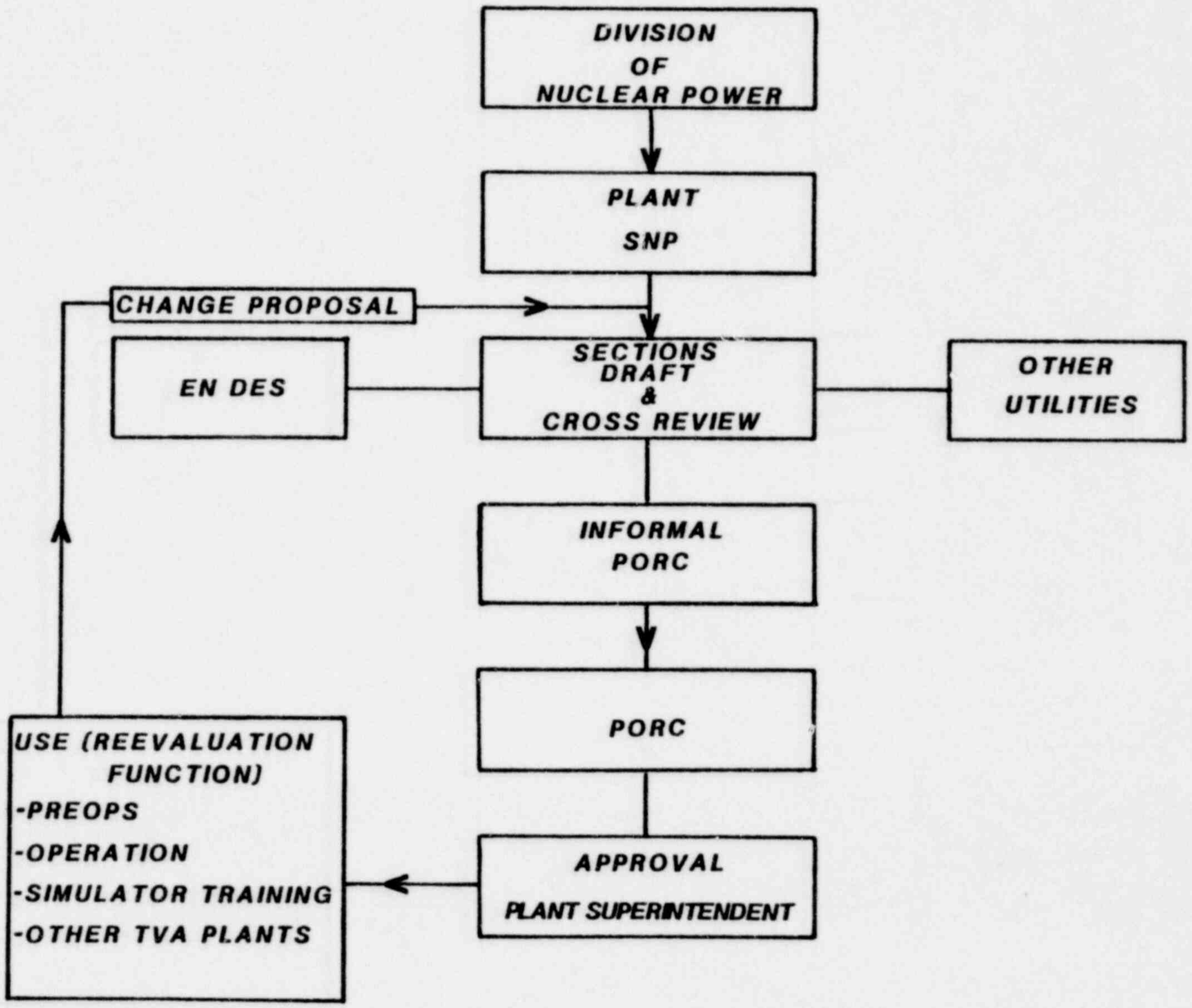


FIGURE A-2.1.9-1

OPERATIONAL PROCEDURES DEVELOPMENT AND REVIEW
SEQUOYAH NUCLEAR PLANT

Figure A-2.1.9-2



70c

Jan. 11, 1980

1759 131

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.

Revisions to the Sequoyah FSAR to include descriptions of the hydrogen analyzer, sampling points, readout and system capabilities will be made by Amendment 64. Refer to pages 74a and 74b for revised FSAR pages.

CLARIFICATION ITEMS

1. The hydrogen analyzers of Sequoyah meet the applicable requirements for qualification, redundancy, and testability in accordance with Sequoyah's commitment to IEEE 323-71.

2. These analyzers are installed and operational.

JAN 11 1980

1759 132

The hydrogen purge exhaust subsystem consists of a single penetration (X-80, Reference Table 6.2-19) in the primary containment wall equipped with two normally closed, remote manually operated isolation valves, one on either side of the containment wall; one pneumatically operated annulus purge exhaust valve located within the annulus; and two 1/2-inch leakoff nipples located between the outboard isolation valve and the annulus purge exhaust valve. With the containment isolation valves open, and the annulus purge exhaust valve closed, a flow path is established from the primary containment through the leakoffs and into the annulus, which will permit purging of the containment for hydrogen control subsequent to a LOCA. The impetus for flow will be provided by the differential pressure between the primary containment and annulus. If the concentration cannot be maintained below 4% through the leakoff path, the annulus purge valve will be opened to supply dilutant air for a minimum time sufficient to maintain the hydrogen concentration below 4%. The containment effluent purged for hydrogen will flow directly to the annulus where it will mix with the annulus atmosphere and be filtered by the air cleanup system prior to discharge to the outside environment. The calculated radiological consequences of the LOCA, including the hydrogen purge, will not exceed the guidelines of 10 CFR Part 100 under the most severe condition in which the annulus purge valve is opened at a pressure differential of 1.0 psid. Redundancy is not required for this system since it is a backup system to the redundant hydrogen recombiners. A piping and instrumentation drawing is provided in Figure 6.2-96a. 37 The flow resistance coefficient (K) used for flow through each 1/2-inch nipple was assumed to be 1.5.

The dilution air flow is introduced into the containment from the service air system. This service air system has provisions enabling it to receive diesel power. The dilution supply subsystem consist of a single 2-inch penetration (X-40D, Reference Table 6.2-19) in the primary containment wall equipped with provisions for containment isolation. The inboard containment isolation feature is a check valve located in the primary containment. The outboard containment isolation feature is a double O-ring sealed flange located in the auxiliary building. A pressure hose will be required to provide a flow path from a service air flange. Operation of this subsystem is accomplished by removing the flange and coupling a service air hose to the pipe penetration. The system will be sized to provide 60 scfm of dilution air at a service air pressure of 60 psig.

The Hydrogen Sampling System is designed to continuously monitor hydrogen concentration inside containment during an accident. The hydrogen analyzers are located in the reactor building annulus and monitor the containment through stainless steel tubing coming from one point in the upper compartment and one point in the lower compartment. These lines are equipped with normally closed, air operated, remote manual isolation valves. The return line is also stainless steel and is equipped with isolation valves identical to those on the incoming lines. Because the analyzers are in the annulus, the 64 accident environment for them is a temperature of 150°F and a radiation dose of 5×10^7 rads. The analyzer internals are designed to process containment atmosphere at 56 psig, 300°F, and 100 percent relative humidity. Hand switches, indicators, and alarms are located in the main control room. The analyzer electronics are located in the auxiliary building. The system is seismically qualified.

When the system is actuated, containment atmosphere is continuously drawn through a series of sample conditioners before entering the analyzer including a trap, moisture separator, and filter. The atmosphere from the upper and lower compartments is mixed before entering the analyzer. As a result of the analyzer capability and the mixing afforded by the hydrogen collection system which draws from compartments within the containment and the containment dome, a true indication will be given of the hydrogen concentration within containment. The analyzers are calibrated to measure hydrogen concentrations between zero and ten percent with an accuracy of plus or minus one-tenth of one percent. This range is sufficient to measure hydrogen releases from metal-water reactions of up to 40 percent. 64

6.2.5.3 Design Evaluation

Prediction of hydrogen generation following the loss-of-coolant accident shows that although hydrogen production rate decreases as time after the accident increases, total hydrogen accumulation can exceed the lower flammability limit of 4 volume percent. Therefore, control measures are necessary to prevent hydrogen accumulation to this limit. The electric recombiner provides the means to prevent unsafe levels of hydrogen concentration from being reached in the containment following a loss-of-coolant accident.

For the purpose of showing that the electric recombiner is capable of maintaining safe hydrogen concentrations, analysis was performed using the AEC Regulatory Guide 1.7 Model. The Regulatory Guide 1.7 Model is based upon assuming a fission product activity release specified in TID-14344 and the values for post-accident hydrogen generation specified in the guide.

Figure 6.2-96 shows the containment hydrogen with one recombiner unit started 24 hours after a loss-of-coolant accident.

Each electric recombiner is capable of continually processing a minimum of 100 scfm of containment atmosphere. Substantially all of the hydrogen contained in the processed atmosphere is converted to steam, thus reducing the overall containment hydrogen concentration. The hydrogen concentration in the containment was calculated for the model described above based on a recombiner capability of 100 scfm of containment atmosphere. This calculation shows that the maximum hydrogen concentration will be less than the lower flammability limit of 4 volume percent if the recombiner is started one day following the accident. Therefore, one of these units meets the design criterion of maintaining a safe hydrogen concentration with considerable margin, and the second unit provides the redundance of a system of equal capability on a redundant power supply.

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 was submitted for NRC review by letter from L. M. Mills to L. S. Rubenstein dated January 11, 1[illegible]

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) Currently, there are no procedures for removal of noncondensible gas from the U-tube regions in the steam generators. WCAP 9600 shows that there will not be a significant accumulation of noncondensible gas in the U-tube region. Small amounts of noncondensible gas that does accumulate in the U-tube region will be removed by natural circulation. The benefits of and necessity for procedures for removal of large amounts of noncondensible gas from the U-tube region will be addressed as part of the Item 2.1.9 task and procedures, if appropriate, will be provided.

   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.

JAN 11 1980

1759 135

Refer to letter from L. M. Mills to L. S. Rubenstein dated January 11, 1980, for the proprietary version of this figure.

FLOW DIAGRAM OF REACTOR VESSEL HEAD VENT SYSTEM
Figure 2.1.9(d)-1

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

NOV 21 1979

1754 137

ADDENDUM TO ITEM 2.2.2.b

SAFETY REVIEW GROUP DURING LOW POWER PHYSICS TESTING PROGRAM

In response to TVA's commitment to provide a separate safety review group during the special low-power physics testing program, the following section was included in Appendix A of the Sequoyah Nuclear Plant OQAM, Part II, Section 4.2:

Central Office Observers

Engineering personnel who are not a part of the plant staff (and do not have direct test responsibility) shall be on each day shift during the special test program to act as independent observers. The duties and responsibilities of these engineers are as follows:

Be cognizant of the scope and intent of the special test program.

Be cognizant of the test being conducted.

Be familiar with the operation of a PWR-type reactor.

Provide daily status reports to the Assistant Director of Nuclear Power (Operations). These reports shall be made by the engineer designated as lead observer on each day shift.

1759 138

JAN 11 1980