50-327

Q

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

July 12, 1979

Mr. Dominic B. 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 No. 50-327
Tennessee Valley Authority	)	

Enclosed are TVA's responses to the requests for additional information transmitted in S. A. Varga's letter to H. G. Parris dated June 1, 1979, concerning Sequoyah Nuclear Plant (SNP) Unit 1.

Enclosure 1 provides responses to each of the 13 items of the NRC-OIE Bulletin 79-06A. These responses to the bulletin are essentially the same as presented to the Advisory Committee on Reactor Safeguards (ACRS) on May 11, 1979. Enclosure 2 is our responses to the ACRS recommendations of April 7, 17, and 20, 1979, which were also presented to the ACRS on May 11, 1979.

As seen from the enclosures, our responses indicate an extensive evaluation of the Three Mile Island (TMI) incident as it relates to the SNP facility design and operation. This evaluation has verified the adequacy of current design and procedural controls governing operation, testing, and emergency response for the Sequoyah facility. However, our evaluation has also identified appropriate changes to improve the nuclear safety and reliability of the Sequoyah facility. These changes in design, testing, operation, and emergency procedures are expressed as

B001
SE
1/1

7907200376

Q

commitments in the enclosures and will be documented in the SNP Final Safety Analysis Report (FSAR). TVA will continue to incorporate lessons learned from its review of the TMI incident into its nuclear plants, both operating and under construction.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosures

ENCLOSURE 1

RESPONSE TO NRC OIE BULLETIN 79-06A

Response to NRC OIE Bulletin 79-06A

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

   c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

Response to question 1

The plant superintendent of TVA's Bellefonte Nuclear Plant, a 2-unit Babcock and Wilcox type reactor plant, presented a review of the TMI incident to all nuclear plant managers and supervisors with operational responsibilities. This review began at Browns Ferry on April 23, 1979, and at Sequoyah Nuclear Plant on May 4, 1979. In addition to studying NRC reports and analysis of the event, he was on site at the Three Mile Island plant in an assistance role during the period of April 9-13, 1979, and is familiar with the circumstances of the TMI incident. Plant management subsequently conducted a comparable review using the same material and training aids for all licensed operators. This review was directed toward (1) understanding the serious consequences of improper alignment of critical systems, (2) evaluating operational actions which could potentially lead to core damage, (3) recognizing the potential for erroneous conclusions based upon a single observation of a given plant parameter, and (4) understanding the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action. Instruction and caution was given to licensed operating personnel in (1) not overriding automatic action of engineered safety features and (2) not making operational decisions on a single observation of a given plant parameter when one or more confirmatory indications are available. The contents of and participation in this review was documented.

Action

2. Review the actions required by your operating procedures for coping with transients and accidents with particular attention to:

    a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

Response to question 2a

Abnormal Operating and Emergency Operating Instructions dealing with transients and LOCA situations have been reviewed. In relation to the recognition of the possibility of forming voids in the primary system, these instructions differentiate between situations in which maximum charging and reactor seal injection flow are adequate to maintain pressurizer water level and thus prevent void formation and those situations in which voiding is an expected occurrence.

A caution will be inserted in the appropriate operating instructions dealing with LOCA situations to indicate the potential of forming voids in the primary system. In most cases, the engineered safety features have been designed to cope with voiding, thus plant operating instructions need not make specific reference to specific operator actions in response to voids in the primary system since this would be the expected condition. These instructions do require verification of proper safety system actuation and operation. Specific cautions and requirements are included for the termination of any automatic safety system function.

There are two situations where void formation is not expected during a loss of coolant event which must be recognized by the operators and appropriate action taken. (1) If the loss of coolant is caused by an open pressurizer relief valve which closes or is isolated before the system depressurizes to hot leg saturation, voiding is not expected. The detection of this type of transient is covered by an abnormal operating instruction which instructs the operator to isolate the faulty relief valve and, if possible, to recover pressurizer pressure and level to their normal operating values. (2) Voiding in the primary system is not expected for breaks sufficiently small such that RCS pressure equilibrates above hot leg saturation when safety injection flow equals break flow. This small LOCA situation is covered by an operating instruction which instructs the operator to maintain RCS pressure by supplying adequate charging flow and to recover pressurizer level to normal if possible. In these two specific cases, confirmation of no voids in the primary system will be by sufficient pressurizer pressure for existing hot leg temperature.

Action

2b. Operator action required to prevent the formation of such voids.

Response to question 2b

As mentioned in the response to Action 2a, accident procedures dealing with LOCA situations instruct the operator to take actions to provide core cooling and to maintain pressurizer pressure above the corresponding hot leg saturation temperature. Examples of immediate required operator actions from these instructions which tend to prevent void formation in LOCA situations include:

1. Verifying that reactor trip has occurred, and that safety injection has been initiated if reactor coolant pressure is below the setpoint.

2. Verifying that residual heat is being dissipated through the steam generators and reactor coolant temperature is stable or decreasing,

3. Verifying that feedwater is being supplied to the steam generators and an indicated level is being maintained in all steam generators not directly affected, and

4. Operator action should be taken to maintain pressurizer water level and pressure by charging and emergency makeup control.

For some LOCA cases, no operator action will prevent the formation of voids in the primary coolant system. The engineered safety features were designed to recover and cool the core following various degrees of primary system voiding, depending on break size and location. In the event of a steam generator tube leak, an operating instruction indicates that if the leak rate is low enough, then charging flow will maintain system pressure above saturation; if not, the system will start to void. The operator has been trained to identify and isolate the faulty steam generator as quickly as possible to prevent or minimize voiding. The procedure for isolating the faulty steam generator is also included in the operating instruction.

Action

2c. Operator action to enhance core cooling in the event such voids are formed (e.g., remote venting).

Response to question 2c

Plant procedures describe the necessary operator actions to ensure core cooling if the primary system is voided. These procedures require operator verification of ECCS component performance and provide for hot leg RHRS injection for enhanced core cooling at a later time in the cooldown process. Plant instructions will contain requirements for operation of the reactor coolant pumps under abnormal conditions to enhance core cooling. Currently, there are no provisions for remote venting of the reactor vessel head. TVA will proceed with the design and installation of this capability for Sequoyah. A design effort has been initiated and as soon as design details are available, they will be submitted for NRC review.

Action

3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

Response to question 3

In the interim, the low pressurizer level setpoint bistables will be tripped to permit safety injection on low pressurizer pressure. Each low level bistable will be removed from the tripped mode only to permit surveillance testing of the coincident low pressurizer pressure bistable. TVA is initiating a design change to the protective logic that will cause initiation of safety injection on 2 out of 3 low pressurizer pressure signals regardless of pressurizer level. This change will be made before fuel loading.

TVA will revise all applicable instructions to require manual initiation of safety injection when two of the three pressurizer pressure signals reach the actuation setpoint.

Action

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Response to question 4

The containment isolation systems function to isolate all non-safety-related fluid systems penetrating the containment upon receipt of a phase A or phase B containment isolation signal. Phase A isolates all process lines except safety injection, containment spray, portions of component cooling, and essential raw cooling water. Phase A isolation can be initiated manually and is initiated by automatic or manual safety injection actuation. Phase B isolates all remaining process lines except safety injection, containment spray, and auxiliary feedwater. Phase B is initiated by 2 out of 4 HI HI containment pressure signal or by manual actuation of containment spray. In addition, isolation valves in the primary containment ventilation system receive a containment ventilation isolation signal. This signal provides for automatic isolation on high radiation and safety injection. See table 1 for a description of containment isolation signals.

Containment isolation does not automatically reset by elimination or resetting of the actuation signal. For example, resetting safety injection will not clear containment isolation; the isolation signal can only be cleared by manual actions on the main control board.

Control features are provided for the containment isolation valves such that:

1. The valves will remain in the closed position if the containment isolation signal is reset.

2. The containment isolation signals override all other automatic control signals.

3. Each valve can be opened or closed manually after the containment isolation signals are reset.

TABLE 1

Safety Injection System (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 (low level bistables tripped)
- 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 least two of the four loops must reach the instrument setpoints to initiate the SIS.

Phase A Initiation

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

Phase B Initiation

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

Containment Ventilation Isolation Initiation

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

Action

5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

Response to question 5

Auxiliary feedwater system is automatically initiated on safety injection, loss of both turbine-driven feed pumps, loss of a single main feed pump coincident with reactor power above 80%, 2 out of 3 low-low steam generator level on any steam generator, or station blackout. This system utilizes two electric motor-driven pumps and one turbine-driven pump. Each motor-driven pump has a capacity of 440 gallons per minute which is sufficient for safe cooldown. The pumps are connected to separate emergency power buses. The turbine-driven pump has a capacity of 880 gallons per minute; steam supply to the turbine is taken from two of four main steam lines at a point upstream of the main steam isolation valves.

Action

6. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power-operated relief valve(s) are open.

Response to question 6a

Plant indications for detecting an open pressurizer relief valve are: (a) control room valve position indication is directly determined from stem mounted switches, (b) inline temperature element downstream of the relief valves with control room temperature indication and high temperature alarm, (c) pressurizer relief tank (PRT) temperature indication and high temperature alarm in the control room, (d) PRT pressure indication and high pressure alarm in the control room, (e) PRT level indication and high level alarm in the control room, and (f) pressurizer pressure indication and low pressurizer pressure alarm in the control room.

Action

6b. Direct the plant operators to manually close the power-operated relief block valve(s) when reactor coolant system pressure is reduced below the setpoint for normal automatic closure of the power-operated relief valve(s) and the valve(s) remain stuck open.

Response to question 6b

A plant Abnormal Operating Instruction is currently available which instructs plant operators to utilize the plant indicators described in 6a above to detect an open pressurizer relief valve. The operator is further instructed to isolate the open valve(s) if they fail to automatically reclose at the proper system pressure.

Action

7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity, then the HPI should be secured (as noted in b(2) below).

Response to question 7a

The operating procedures and training instructions are being reviewed to ensure that operators are instructed not to override automatic operations of the engineered safety features, unless continued operation of the engineered safety system will result in unsafe plant conditions, or until the plant is clearly in a stable, controlled state, and engineered safeguards are no longer required. This review will be completed before the fuel loading of unit 1.

Action

7b. Operating procedures currently, or are revised to, specify that if the high-pressure injection (HPI) system has been automatically actuated because of low-pressure conditions, it must remain in operation until either:

   (1) Both low-pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer at a rate which would ensure stable plant behavior, or

   (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees Fahrenheit and the length of time HPI is in operation shall be limited by the pressure/temperature consideration for the vessel integrity.

Response to question 7b

The commitment expressed in the response to question 7a guarantees adequate run time for both low-pressure and high-pressure injection pumps. However, applicable operating instructions, abnormal operating instructions, and emergency operating instructions will be revised to require operation of high-pressure injection pumps for 20 minutes following automatic actuation unless all of the following conditions are met: (a) reactor coolant system pressure above safety injection setpoint, (b) reactor coolant pressure stable or increasing, (c) at least one steam generator available for primary system cooling, and (d)pressurizer level above the point at which pressurizer heater can be utilized. These conditions will ensure that the plant is in a controlled state with an excess of 50°F subcooling based on Tave. If the above

conditions are met, HPI can be shutoff and subcooling will be monitored by calculating the saturation temperature corresponding to the measured pressurizer pressure. This saturation temperature will be compared to the hot leg temperatures. If 50 degrees subcooling cannot be maintained after HPI shutoff, then HPI will be reactivated. The degree of subcooling beyond 50 degrees Fahrenheit and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for vessel integrity.

Action

7c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating for two loop plants and at least two RCP's shall remain operating for 3 or 4 loop plants as long as the pump(s) are providing forced flow.

Response to question 7c

The Sequoyah NSSS vendor, Westinghouse, has not fully evaluated the above NRC recommendation, and Westinghouse continues to recommend that all reactor coolant pumps (RCP's) be tripped following steam break and loss of coolant accidents. Since TVA has no technical basis for disputing the Westinghouse recommendation, Sequoyah operating instructions will specify the conditions under which the pumps should be tripped based on Westinghouse guidelines.

If Westinghouse's full evaluation of NRC's recommendation shows the acceptability of maintaining at least two reactor coolant pumps in operation following high pressure injection, Sequoyah operating procedures will be revised to require operation of at least two RCP's provided pressurizer level indication is above zero, specific control room instrumentation (motor amps, loop flow, loop temperature, and loop pressure) clearly indicate that the RCP's are providing stable forced flow, and a Phase B containment isolation signal is not present. As discussed in the response to question 4, Phase B isolation will isolate the component cooling system blocking coolant flow to the reactor coolant pumps.

Action

7d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

Response to question 7d

TVA's training program emphasizes the interpretation of all available information in order that the operator can diagnose the basic cause of any malfunction or abnormal occurrence. The plant abnormal and emergency instructions list specific confirmatory indications and expected system parameter changes associated with equipment malfunctions or postulated accidents. The prescribed operator response to the abnormal situations also lists the confirmatory indications to verify appropriate corrective action is being taken.

In addition to pressurizer level, there are other types of instrumentation that will provide the operator with indirect indications of primary system coolant inventory changes and could inform the operator of the need to take corrective action. Examples are listed below.

Primary system pressure
Primary system temperature
Containment pressure and temperature
Containment radiation levels
Relief valve tailpipe temperature
Pressurizer relief tank level, temperature, and pressure
Reactor coolant pump motor amps
Containment moisture alarms
Containment sump level indications and alarms
Ice condenser doors opening alarm
Ice bed temperature alarms
Charging flow

Action

8. Review all safety-related valve positions, positioning requirements, and positive controls to ensure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Response to question 8

Positioning requirements for safety-related valves will be reviewed for correctness and completeness for all operational modes to ensure the proper operation of engineered safety features. Additionally, system operating instructions, maintenance instructions, test instructions, and surveillance instructions will be reviewed to ensure proper positioning of these valves. This review and any necessary modifications will be completed before the fuel loading unit 1.

Current plant administrative procedures require that (a) all essential safety system and component alignment is verified prior to unit startup, (b) changes in the alignment of any safety system component is recorded on a system status sheet, and (c) shift personnel being relieved communicate information on any abnormal plant condition including temporary conditions.

Plant operating instructions require completion of a prestartup checklist prior to unit startup. This checklist is used to verify correct alignment of all safety systems. Alignment of critical systems is reviewed on a weekly basis. Anytime a critical component is changed from its normal position or condition, a system status sheet is completed and placed in a system status folder. In addition, panel checklists are reviewed weekly to verify that proper panel alignment exists for all safety systems.

Return of a system or component to its normal mode or status following maintenance or testing is addressed in the response to action 10.

Action

9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to ensure that undesired pumping, venting, or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is ensured.

Response to question 9

We have reviewed the operating modes and procedures for all systems designed to transfer potentially radioactive liquids and gases from the primary containment. Those systems or components include the reactor building purge ventilating system, waste disposal system, the containment building floor and equipment drain sump pumps, and the reactor coolant drain tank pumps and vent line.

The function of the containment floor and equipment drain sump is to collect and measure nuclear system leakage from both identified and unidentified sources. Two sump pumps operate automatically to maintain the sump level within a desired range; the pumped fluid is transferred to the waste disposal system. The reactor coolant drain tank collects reusable reactor coolant water from inside the containment (excess letdown flow, leakoff flow, pressurizer relief tank drains, etc.). Two pumps are available to transfer the fluid through isolation valves to the chemical and volume control system holdup tanks.

The isolation valves on the discharge lines from the floor and equipment drain sump and the reactor coolant drain tank (RCDT) auto-close on initiation of safety injection or Phase A isolation. In addition, the vent line from the RCDT to the waste disposal system vent header is equipped with two in series containment isolation valves that auto-close on safety injection or Phase A isolation. All of the above valves will remain closed following reset of safety injection or Phase A isolation. Manual operator action is required to open each valve. In addition, these valves are designed to fail closed. There are currently no high radiation signal(s) to close any of these valves.

The lines in the waste disposal system isolate automatically upon actuation of the SIS. Resetting the SIS initiation does not permit any containment isolation valve to reopen. The operator must reopen individually the two isolation valves in each line after resetting the SIS signal before any fluid can be transferred out of containment.

The reactor building purge system is designed to supply fresh air for breathing and contamination control to allow personnel access for maintenance and refueling operations. Each purge system containment penetration is provided with both inboard and outboard motor-operated isolation butterfly valves. Containment ventilation isolation signal automatically shuts down the purge air supply fans and closes their discharge dampers and butterfly valves. Containment ventilation isolation is generated by high radiation in the purge exhaust line, high radiation in the upper and lower containment (gaseous and particulate radiation monitors), safety injection, and manual Phase A or Phase B isolation. Isolation valves will remain closed following reset of containment ventilation isolation. Manual operator action is required to open each valve. In addition, these butterfly valves are designed to fail closed.

TVA will proceed with the design and installation of radiation detectors for Sequoyah which will automatically isolate the RCDT and the floor and equipment drain surge when high radiation is detected. As soon as design details are available, they will be submitted for NRC review.

Operability of the above features are ensured through surveillance testing of the applicable components as required by plant technical specifications.

Action

10. Review and modify, as necessary, your maintenance and test procedures to ensure that they require:

    a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

    b. Verification of the operability of all safety-related systems testing.

    c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Response to question 10

Current plant administrative procedures: (a) require verification of the operability of redundant safety-related equipment before such equipment is removed from service (equipment operability requirements are based on plant technical specification), (b) require that system operability is demonstrated before a system is returned to service, and (c) require approval by the shift supervisor or his representative prior to the performance of any activity on safety-related plant equipment, or any activity that may affect safety-related plant equipment. In addition, the shift supervisor or his representative is notified when an activity authorized to be performed on safety-related plant equipment is completed or a change occurs in the scope of the activity.

Detailed plant maintenance and test procedures will be reviewed, and any found not meeting the above requirements will be modified before fuel loading of unit 1.

In addition to prewritten maintenance instructions (MI's), maintenance activities are controlled through a maintenance request (MR) system which identifies specific maintenance requirements on each MR. Before maintenance is performed, the work requirements of the MR are reviewed by the operations superintendent or his representative. Approval to perform maintenance on safety-related equipment is indicated by signature of the shift supervisor on each MR is required by plant administrative procedures. After completion of work, the shift supervisor or his representative is notified that the maintenance activity is complete. Return to normal instructions and test requirements are specified in a referenced MI, surveillance instruction, or on the initiating MR.

Normal hydrogen levels in the primary system are accommodated by the plant chemical volume control system. Letdown flow to the volume control tank permits primary system hydrogen purging to the waste gas system. The partial pressure of hydrogen in the volume control tank controls the hydrogen concentration in the primary makeup water. The water is returned to the primary loop through the charging pumps.

Hydrogen may accumulate in the primary containment following a loss-of-coolant accident. This hydrogen is controlled via the containment combustible gas control system; this system is composed of two redundant hydrogen recombiner units permanently located in the upper containment compartment. A redundant hydrogen sampling system qualified to process the post-LOCA atmosphere is used to provide control room indication of containment hydrogen concentration. Each recombiner is sized to limit hydrogen concentrations below 4 percent which is the accepted lower flammability limit for hydrogen. Adequate mixing of the containment atmosphere is provided by the containment air return fan system. Post-LOCA hydrogen mixing capability is provided by the air return fan system in the following regions of containment: containment dome, each of the steam generator enclosures, pressurizer enclosure, upper reactor cavity, each of the accumulator rooms and the instrument room. This ensures optimum recombiner action by preventing the local concentration of hydrogen. A remote manual hydrogen purge system is also provided to limit the flammable gas concentration to 4 percent in the absence of recombiner action. The containment air space is purged to the annulus and replenished by a dilution air supply. The air entering the annulus will mix with the annulus air and is processed by the emergency gas treatment system before discharge to the outside environment. This system provides a backup for the recombiner system.

Existing plant procedures for controlling containment hydrogen concentration using the above procedures have been reviewed and found acceptable.

Action

11. Review your prompt reporting procedures for NRC notification to ensure that NRC is notified within one hour of the time the reactor is not in a controlled or unexpected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

Response to question 11

This requirement will be incorporated into our Sequoyah Nuclear Plant Procedures.

Action

12. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

Response to question 12

The methods for removing hydrogen from the reactor coolant system are:

1. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if a reactor coolant pump is operating in a loop from which pressurizer spray is provided.

2. Hydrogen in the pressurizer vapor space can be vented by power-operated relief valves to the pressurizer relief tank.

3. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system. The waste gas system consists of 9 tanks of 600 ft^3 each at a maximum of 100 psig.

4. In the event of a LOCA, hydrogen would vent with the steam to the containment.

If a noncondensable gas bubble becomes situated in the primary coolant system, there are many options for continued core cooling and removing the bubble.

With a gas bubble located in the upper head, several methods of core cooling are unaffected. The steam generators can be used to remove decay heat using reactor coolant pump forced flow or natural circulation. The safety injection system can be used to cool the core while venting through the pressurizer power-operated relief valve. Core cooling by either of these methods can proceed indefinitely if the primary coolant pressure is held constant. If a lower system pressure is desired, a controlled depressurization will allow the bubble to grow slowly until it uncovers the top of the hot leg and is expelled through the pressurizer power-operated relief valve.

Existing plant procedures dealing with LOCA situations will be revised to include instructions to the operator for dealing with a noncondensable gas bubble in the primary system based upon the methods described above before fuel loading of unit 1.

Normal hydrogen levels in the primary system are accommodated by the plant chemical volume control system. Letdown flow to the volume control tank permits primary system hydrogen purging to the waste gas system. The partial pressure of hydrogen in the volume control tank controls the hydrogen concentration in the primary makeup water. The water is returned to the primary loop through the charging pumps.

Hydrogen may accumulate in the primary containment following a loss-of-coolant accident. This hydrogen is controlled via the containment combustible gas control system; this system is composed of two redundant hydrogen recombiner units permanently located in the upper containment compartment. A redundant hydrogen sampling system qualified to process the post-LOCA atmosphere is used to provide control room indication of containment hydrogen concentration. Each recombiner is sized to limit hydrogen concentrations below 4 percent which is the accepted lower flammability limit for hydrogen. Adequate mixing of the containment atmosphere is provided by the containment air return fan system. Post-LOCA hydrogen mixing capability is provided by the air return fan system in the following regions of containment: containment dome, each of the steam generator enclosures, pressurizer enclosure, upper reactor cavity, each of the accumulator rooms and the instrument room. This ensures optimum recombiner action by preventing the local concentration of hydrogen. A remote manual hydrogen purge system is also provided to limit the flammable gas concentration to 4 percent in the absence of recombiner action. The containment air space is purged to the annulus and replenished by a dilution air supply. The air entering the annulus will mix with the annulus air and is processed by the emergency gas treatment system before discharge to the outside environment. This system provides a backup for the recombiner system.

Existing plant procedures for controlling containment hydrogen concentration using the above procedures have been reviewed and found acceptable.

Action

13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

Response to question 13

A technical specification change has been submitted to reflect tripping the pressurizer level bistables. Technical specifications will again be revised when the logic change discussed in the response to question 4 is implemented. Other changes will be submitted for staff review if upon completion of our detailed evaluation, further modifications are considered necessary.

ENCLOSURE 2

TVA EVALUATION OF GENERIC ACRS QUESTIONS

ACRS Statement

It would be prudent to consider expeditiously the provision of instrumentation that will be providing an unambiguous indication of the level of fluid in the reactor vessel.

Response

To meet the need for better information concerning the level of fluid in the reactor vessel, TVA will provide level measurement instrumentation for the Sequoyah Nuclear Plant. As soon as design details are available, they will be submitted for NRC review.

ACRS Statement

Early consideration should be given also to providing remotely controlled means for venting high points in the reactor system, as practical.

Response

High points in the SQN reactor coolant system (RCS); i.e., locations where gases and/or vapors have the potential to accumulate, are as follows:

1. Uppermost region of reactor vessel.

2. Uppermost region (U-bend) of each steam generator (four).

3. Uppermost region of pressurizer.

Of the regions listed above, currently only the uppermost region of the pressurizer has remotely controlled means for venting; the controls are located in the auxiliary building, and in the MCR. The vent line is routed to the PRT.

The reactor vessel currently requires local manual operations for venting through a small diameter (3/4-inch diameter) line.

It is not possible to directly vent the U-bend region of a steam generator.

TVA will provide remote venting capability for the reactor vessel at Sequoyah. A careful design effort has been initiated and as soon as design details are available, they will be submitted for NRC review.

ACRS Statement

The committee believes that greater understanding of this mode of cooling (natural circulation) is required and that detailed analyses should be developed by licensees or their suppliers. The analyses should be supported, as necessary, by experiment.

Response

In order to prevent confusion, the definition of natural circulation must be established prior to discussing this phenomena. Natural circulation is a condition in the reactor coolant system (RCS) wherein the RCS fluid is single phase water, no forced circulation of the water exists, but water density differences between the water in the reactor pressure vessel and the steam generators exists such that a driving head across the core results. This definition is apparently consistent with the ACRS definition of natural circulation, but may not be consistent with the NRC definition.

The implications of Three Mile Island (TMI) plus the traditional single failure licensing philosophy have been considered in our evaluation of natural circulation at SQN.

Natural circulation is one of the important modes of decay heat removal during the course of an entire family of loss of coolant accidents (LOCA's) characterized as small break loss of coolant transients. The other modes are heat removal through the break and by steam condensation. Any break in the reactor coolant pressure boundary larger than 0.375 inch I.D. (0.008 sq. ft.) and smaller than 9.57 inches I.D. (0.50 sq. ft.) is categorized as a small break on a Westinghouse pressurized water reactor. The following discussion on natural circulation following a small break loss of coolant transient is based on the latest analyses performed by Westinghouse in light of TMI. The base plant considered in these analyses is a four loop RESSAR-3 plant.

The break size in a small break loss of coolant transient is the determining factor as to whether or not the steam generators are relied upon as a heat sink during the initial portion that is, approximately the first 24 hours of the event. Westinghouse has shown that for breaks 2 inches I.D. (0.022 sq. ft.) and smaller, the steam generators are relied upon as a heat sink during the initial portion of the transient until the break flow is capable of removing decay heat. (Typically for a 1-inch I.D. line break, it would take approximately 24 hours before the break flow can remove decay heat.) For breaks larger than 2 inches I.D., the steam generators are not relied upon as a heat sink during the initial portion of the transient because the break flow is large enough to remove decay heat very early in the event.

Westinghouse has concluded that natural circulation of fluid through the reactor coolant system will not be interrupted during any of the small breaks they have analyzed. Their bases for this conclusion are as follows:

1. There is not a large enough source of noncondensibles during any of the small breaks analyzed which has the potential to bind up the U-tubes in the steam generators.

2. The physical characteristics of the U-tube steam generators used in Westinghouse plants prevent them from being susceptible to noncondensible binding; any steam and noncondensibles that enter the steam generator will pass through an area of the steam generator that is surrounded by a substantial amount of water on the secondary side, causing the steam to condense, and reducing the steam and noncondensible bubble size to the point that it cannot cause binding of the U-tubes in the steam generators.

3. Even if large amounts of noncondensibles were present in the reactor coolant system, Westinghouse has modeled, calculated, and concluded that any noncondensibles that enter the steam generator U-tubes will be swept out due to the inherent differences between the water and noncondensible velocities. Subsequently, buildup of noncondensibles in the high points of the reactor coolant system will be prevented.

It is TVA's conclusion that natural circulation will not be interrupted in a Westinghouse PWR as a result of the formation and/or introduction of non-condensibles during a small break loss of coolant transient. This conclusion is based on TVA's current understanding of natural circulation in a Westinghouse PWR. However, TVA will continue to work with Westinghouse to ensure that both organizations' understanding of natural circulation as a result of small break loss of coolant transients remains valid as the exact nature and implications of TMI evolve.

ACRS Statement

The plant operator should be adequately informed at all times concerning the conditions of reactor coolant system operation which might affect the capability to place the system in the natural circulation mode of operation or to sustain such a mode. Of particular importance is that information which might indicate that the reactor coolant system is approaching the saturation pressure corresponding to the core exit temperature. This impending loss of system overpressure will signal to the operator a possible loss of natural circulation capability. Such a warning may be derived from pressurizer pressure instruments and hot leg temperatures in conjunction with conventional steam tables. A suitable display of this information should be provided to the plant operator at all times.

Response

a. Presently, the Sequoyah process computer monitors four hot leg temperatures (HLT's) and four pressurizer pressures (PP's) and obtains an average of each. The computer programs include steam table conversions. Also, the computer has trend recorders with dual pens.

b. TVA will add program(s) to calculate the saturation temperature corresponding to the measured pressurizer pressure (avg.). We have the capability to trend the HLT (avg. or any leg) on one pen and the calculated saturation temperature on the other pen. The degrees of subcooling can be observed as the difference between the two pens. An alarm function would be added to indicate when the subcooling ΔT is abnormal. The operator could select the points for trend at that time. (The calculation would be performed every 64 seconds.)

c. TVA will also have steam tables and/or saturation curves available to the control room operator at all times.

ACRS Statement

The exit temperature of coolant from the core is currently measured by thermocouples in many PWR's to determine core performance. The Committee recommends that these temperature measurements, as currently available, be used to guide the operator concerning core status. The range of the information displayed and recorded should include the full capability of the thermocouples. It is also recommended that other existing instrumentation be examined for its possible use in assisting operating action during a transient.

Response

a. Presently, the Sequoyah process computer monitors 65 incore CA (type K) thermocouples. They are now ranged from 0-700°F and calibrated for highest degree of accuracy between 400-700°F (± 3/8 percent). They should be within ±2°F below 400°F.

b. TVA is in the process of changing the software out-of-range index to 1800°F. Accuracy in the upper range will be considerably less than the 0-700°F range (± 20°F). The software change will be complete before Sequoyah unit 1 fuel loading.

ACRS Statement

The use of natural circulation for decay heat removal following a loss of offsite power sources requires the maintenance of a suitable overpressure on the reactor coolant system. This overpressure may be assured by placing the pressurizer heaters on a qualified onsite power source with a suitable arrangement of heaters and power distribution to provide redundant capability.

Response

There are four banks of pressurizer heaters:

1 automatic control group at 415KW
3 backup groups at 485, 485, and 415 KW

Pressurizer low level will trip all four banks of the heaters and prevent them from coming back on until level is recovered in the pressurizer. 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 would come on automatically. The other two backup heater banks and the control bank would not come on automatically, but could be manually activated. All four heater banks can be powered from the on-site emergency diesel generators. The control bank and one backup bank of heaters are supplied from one electrical power train and the other two backup banks are supplied from the other power train.

ACRS Statement

Consideration should be given to the desirability of additional equipment status monitoring on various engineered safeguards features and their supporting services to help assure their availability at all times.

Response

1. The status monitoring system automatically presents the operator in the main control room with a visual display and alarm, indicating the status of any ECCS system which has been deliberately bypassed or deliberately made inoperable. This system meets the conditions described in Section C of Regulatory Guide 1.47.

   The visual display consists of a schematic flow diagram of the bypassed or inoperable system(s), the status of each component to which Section C of RG 1.47 is applicable is indicated on the face of a cathode ray tube. In addition, a clock is provided indicating the time remaining before the system must be returned to normal or the unit shut down as required by technical specifications.

   The SMS does not currently monitor:

   1. Solenoid valves for which the loss of power causes the valve to go to a safe position
   2. Backpressure valves on the motor-driven pump discharges
   3. Manual maintenance valves
   4. Check valves
   5. Auxiliary equipment and support systems

TVA is proceeding to expand the Status Monitoring System capability for the Sequoyah Nuclear Plant. As soon as design details are available, they will be submitted for NRC review.

ACRS Statement

The ACRS recommends that operating power reactors be given priority with regard to the definition and implementation of instrumentation which provides additional information to help diagnose and follow the course of a serious accident. This should include improved sampling procedures under accident conditions and techniques to help provide improved guidance to offsite authorities, should this be needed.

Response

1. The following post accident instrumentation is supplied to enable the operator to follow transients.

   a. T hot or T cold (measured wide range)

   b. Pressurizer water level

   c. RCS pressure (wide range)

   d. Containment pressure

   e. Steam line pressure

   f. Steam generator water level (wide range)

   g. Steam generator water level (narrow range)

   h. RWST water level

   i. Containment water level

   j. Pressurizer pressure

   k. Containment H_2 monitors

Each of the above channels is either recorded or logged.

2. Containment Radioactivity Levels

   a. Airborne radioactivity levels in the primary containment during accident conditions can be indirectly obtained with the high range area monitor that is located outside the upper compartment personnel hatch. This monitor will remain on scale for containment airborne radioactivity concentrations up to about 20 percent of those that could be experienced in a RG 1.4 loss-of-coolant accident.

   There is no provision for direct measurement during accident conditions of exposure rates or nuclide radioactivity concentrations in the primary containment. There are no radiation monitors inside the containment that have sufficient range and atmospheric qualification for the measurement of radiation levels in the containment during accident conditions corresponding to RG 1.4 assumptions.

Under normal conditions, real time detection of airborne particulate, iodine, and gross radioactivity concentrations is provided by two three-channel monitors per reactor unit. For these monitors, samples of containment air are pumped to the detection assemblies which are located in the auxiliary building. After containment isolation, the isolation valves on the sample lines may be manually reopened from the main control room; however, this action cannot be taken until the containment atmospheric conditions permit it since the monitors are not designed to operate with sample pressure, temperature, and humidity conditions that would exist during some accidents. Even after sample pressure, temperature and humidity conditions return to acceptable values, the monitor channels would be offscale for containment activity levels corresponding to RG 1.4 assumptions.

TVA will install a radiation monitor outside of containment capable of monitoring airborne radiation inside containment corresponding to RG 1.4 assumptions. As soon as design details become available they will be submitted for NRC review.

b. Containment Air Sample

Currently, there is no provision to take containment atmospheric samples for laboratory analysis during harsh containment atmospheric pressure, temperature, and humidity conditions. During normal conditions, the monitors referenced in part (a) provide the following samples that can be analyzed in the laboratory: (1) particulate filter, (2) charcoal absorption cartridge, and (3) a gaseous sample. However, the sampling system for these monitors is not qualified for operation when containment atmospheric conditions correspond to RG 1.4 assumptions. Furthermore, were such samples collectable with these monitor assemblies during accident conditions, there is not sufficient radiation protection for personnel to remove the samples and analyze them in the laboratory.

TVA will modify portions of the existing gaseous sampling system so that shielded samples of RG 1.4 containment atmosphere can be taken in an accessible area. As soon as design details are available, they will be submitted for NRC review.

Under accident conditions, the hydrogen content of the containment atmosphere is monitored with two analyzers located in the annulus between the containment and the shield building. Remote indication is provided in the main control room. These analyzers are redundant safety grade and are on trained power.

c. Water Samples

During normal operation, reactor coolant samples, cooled with component cooling water, are available in the hot sample room. During accident conditions, the containment isolation valves on the sample lines can be opened and reactor coolant samples will again be available in the hot sample room. During normal reactor shutdown operations, samples of the reactor coolant water being cooled by the residual heat removal system (RHR) are taken from RHR pipes and routed to the hot sample room. During accident conditions, these samples, which are available in the hot sample room, would be samples of the sump water under the reactor vessel that is being recirculated. The radiation protection design for taking these

samples and analyzing them in the laboratory is based on operation with up to 1.0% failed fuel. The samples could not be taken and analyzed when sample specific activities are even a small fraction of those corresponding to RG 1.4 assumptions.

TVA will make provisions for sampling water from the reactor coolant system (RCS) and the residual heat removal system (RHR) for activities corresponding to RG 1.4 assumptions. The radiation monitor(s) will be placed on the RHR piping to monitor containment sump water activities corresponding to RG 1.4 assumptions. As soon as design details are available, they will be submitted for NRC review.