TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

October 12, 1979

Mr. Domenic 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 Advisory Committee on Reactor Safeguards (ACRS) recommendations listed in their interim reports, Nos. 2 and 3, dated May 16, 1979. We were asked to respond to all ACRS concerns on the Three Mile Island II incident in S. A. Varga's letter to H. G. Parris dated June 1, 1979. TVA previously responded to this request in my letter to you dated July 12, 1979.

Very truly yours,

L. M. Mills

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosures (40)

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RESPONSES TO ACRS CONCERNS

IN INTERIM REPORT NO. 2 ON TMI-2

ACRS Statement--Natural Circulation--Procedures

It is evident from the experience at TMI-2 that there was failure to establish natural circulation of water in the primary system and failure to recognize in a timely manner that natural circulation had not been achieved. The need for natural circulation under certain circumstances is common to all PWR's.

The Committee recommends that procedures be developed by all operators of PWR's for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has in fact been established. These procedures should take into account the behavior of the systems under a variety of abnormal conditions.

As a first step, the NRC Staff should initiate immediately a survey of operating procedures for achieving natural circulation, including the case when offsite power is lost. At the same time, the operators of all PWR plants should be requested to develop detailed analyses of the behavior of their plants following anticipated transients and small breaks in the primary system, with appropriate consideration of potential abnormal conditions, operator errors and failures of equipment, power sources, or instrumentation. These analyses are necessary for the development of suitable operating procedures. The review and evaluation of these analyses by the NRC Staff should receive a priority consistent with the priority being given to changes in operating procedures.

Response

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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 predominantly 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 exist 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 Sequoyah.

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 and other SAR Chapter 15.0 events. The other modes are heat removal through the break and steam condensation in the reflux boiling mode. Any break in the reactor coolant pressure boundary larger than 0.375 inch ID (0.008 sq ft) and smaller than 9.57 inches ID (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 (approximately the first 24 hours) of the event. Westinghouse has shown that for breaks two inches ID (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 ID line break, it would take approximately 24 hours before the break flow can remove decay heat.) For breaks larger than 2 inches ID, 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 as defined above will not be interrupted during small breaks larger than 3/8 inch ID and smaller than 1 inch ID. Their bases for this conclusion are as follows:

1. The system will reach an equilibrium pressure which corresponds to the pressure at which the liquid phase break flow equals the high head centrifugal pump injection rate.

2. This equilibrium pressure will be established below the steam generator safety valve setpoints for these break sizes.

3. The fluid in the reactor coolant system is saturated or subcooled liquid except in the core and hot legs, where small values of void fraction exist.

4. The steam generator tubes do not drain and the natural circulation mode of decay heat removal will continue to function until the time that the break can remove all the decay heat.

5. No core uncovery is predicted to occur for breaks of this size.

For breaks greater than 1 inch ID, Westinghouse has concluded that natural circulation, as defined above, will be interrupted. However, sufficient heat removal capability by way of either a combination of break flow and steam condensation in the steam generators or break flow alone exists such that peak clad temperatures are limited to below 1800°F and the amount and duration of core uncovery are of little consequence to the outcome of the transient.

Westinghouse has concluded that decay heat removal via the steam generators during small break loss-of-coolant accidents with the RCS in the natural circulation or steam condensation (reflux boiling) modes of operation is not susceptible to interruption due to the introduction and/or presence of noncondensibles. 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.

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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 had 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.

   However, TVA will continue to work with Westinghouse to ensure that both organizations' understanding of natural circulation during small break loss-of-coolant accidents and other Chapter 15.0 events remains valid as the understanding and implications of TMI evolve.

   At the present time, TVA supports the Westinghouse conclusions based on TVA's current understanding of a Westinghouse PWR's response during small break loss-of-coolant accidents.

Instructions for initiation of natural circulation, including the case when offsite power is lost, are contained in an existing emergency operating procedure. These instructions describe the expected response of existing control room instrumentation used to verify that natural circulation has been established. In addition, these instructions describe the actions required to enhance natural circulation including those conditions during which saturation temperature and pressure conditions are reached in the primary system.

ACRS Statement--Natural Circulation--Pressurizer Heaters

The use of natural circulation for decay heat removal following an accident in a PWR normally requires the maintenance of a suitable over-pressure on the reactor coolant system in order to prevent the generation of steam which can impede circulation. For many transients, maintenance of this overpressure is best accomplished by use of the pressurizer heaters.

Although the pressurizer heaters at TMI-2 continued to receive power from offsite sources during the entire accident, the availability of offsite power cannot be assured for all transients or accidents during which, or following which, natural circulation must be established. The Committee recommends that the NRC Staff initiate immediately a survey of all PWRs licensed for operation to determine whether the pressurizer heaters are now or can be supplied with power from qualified onsite sources with suitable redundancy.

Response

There are four banks of pressurizer heaters:

1 automatic control group at 415 kW
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 operate automatically. The other two backup heater banks and the control bank would not come on automatically, but could be manually activated. In the event of a loss of offsite power and safety injection, two backup heater banks rated at 485 kW each and powered from different trains of emergency power can be manually activated from the main control room 90 seconds after emergency power becomes available.

ACRS Statement--Natural Circulation--Saturation Conditions

The plant operators should be informed adequately at all times of those conditions in the reactor coolant system that might affect their capability to place the system in the natural circulation mode or to sustain it in such a mode. Information indicating that coolant pressure is approaching the saturation pressure corresponding to the core exit temperature would be especially useful, since an impending loss of overpressure would signal to the operator a potential loss of natural circulation. This information can be derived from available pressurizer pressure and hot leg temperature measurements, in conjunction with conventional steam tables.

The Committee recommends that information for detecting an approach to saturation pressure be displayed to the operator in a suitable form at all times. Since there may be several equally acceptable means of providing this information, there is no need for the NRC Staff to assign a high priority to the development of prescriptive requirements for such displays. However, a reasonably early request that licensees and vendors consider and comment on the need for such a display would be appropriate.

Response

1. 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.

2. 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.)

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

ACRS Statement--Core Exit Thermocouples

The NRC Staff should request licensees and vendors to consider whether the core exit temperature measurements might be utilized, where available, to provide additional indication regarding natural circulation or the status of the core. For the latter purpose it is recommended that the full temperature range of the core exit thermocouples be utilized. At TMI-2, the temperatures displayed and recorded did not include the full range of the thermocouples.

The Committee believes it would be appropriate for the NRC Staff to request licensees and vendors to consider and comment on this recommendation. This request should be made as soon as convenient and the time allowed for responses should be such as not to degrade responses on higher priority matters. Plant changes that might result eventually from consideration of this recommendation would not, at this time, seem to require a high priority.

Response

1. 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.

2. 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--Instrumentation to Follow the Course of an Accident

The ability to follow and predict the course of an accident is essential for its mitigation and for the provision of credible and reliable predictions of potential offsite consequences. Instrumentation to follow the course of an accident in power reactors of all types has long been a concern of the ACRS, is the subject of Regulatory Guide 1.97 (which has not yet been implemented on an operating plant), and is the subject of an NRC Staff Task Action Plan for the resolution of generic issues.

The Committee believes that the positions of Regulatory Guide 1.97 should be reviewed, and redefined as necessary, and that the Task Action Plan should be reexamined, as soon as manpower is available. The lessons learned from TMI-2 should be the bases for these reviews. For example, improved sampling procedures under accident conditions should be considered.

Although revie nd reexamination of existing criteria may take some time, the studies completed to date, together with the understanding gained from the accident at TMI-2, should provide sufficient basis for planned and appropriately phased actions. The Committee believes that the installation of improved instrumentation on operating reactors of all types should be underway within one year.

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 perce..t of those that could be experienced in a RG 1.4 loss-of-coolant acci nt.

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 3-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 tmospheric 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 acitivity levels corresponding to RG 1.4 assumptions.

TVA will in all redundant radiation monitors 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 percent 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.

ACRS Statement--Reactor Safety Research

The ACRS recommends that safety research on the behavior of light-water reactors during anomalous transients be initiated as soon as possible and be assigned a high priority. The ACRS would expect to see plans and proposals within about three months, preliminary results within an additional six months, and more comprehensive results within a year.

Of particular interest would be the development of the capability to simulate a wide range of postulated transient or accident conditions, including various abnormal or low probability mechanical failures, electrical failures, or human errors, in order to gain increased insight into measures that can be taken to improve safety.

The new program of research to improve reactor safety has been initiated only recently, and then only on a relatively small scale. The Committee reiterates its previous recommendations that this program be pursued and its expansion sought by the Commission with a greater sense of urgency.

Response

Not applicable.

ACRS Statement--Status Monitoring

Although the closed auxiliary feedwater system valves may not have contributed directly or significantly to the core damage or environmental releases at TMI-2, the potentially much more severe consequences of unavailability of engineered safety features in plants of any type is of concern and deserving of attention. Status monitoring not dependent chiefly on administrative control, and thus possibly less subject to human error, might help assure the availability of essential features.

A request should be made within the next few months that licensees consider additional status monitoring of various engineered safety features and their supporting services. The NRC Staff should begin studies on the advantages and disadvantages of such monitoring on about the same time scale. Responses from licensees should be expected in about one year, at which time the NRC Staff should be in a position to review and evaluate them.

Response

The status monitoring system automatically presents the operator in the main control room with a visual display and ala[illegible] 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.

RESPONSES TO ACRS CONCERNS

IN INTERIM REPORT NO. 3 ON TMI-2

1157 014

ACRS Statement--Reactor Pressure Vessel Level Indication

The Committee believes that it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. We suggest that licensees of all pressurized water reactors be requested to submit design proposals and schedules for accomplishing this action. This would assure the timely availability of reviewed designs if the Staff ongoing studies should indicate that early implementation is required. The Committee believes that as a minimum, the level indication should range from the bottom of the hot leg piping to the reactor vessel flange area.

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--Operator Training and Qualification

The NRC Staff should examine operator qualifications, training, and licensing to determine what changes are needed. Consideration should be given to educational background, to training methods, and to content of the training program. Attention should also be given to testing methods, with specific concern for the ability of the testing methods to predict operator capability. Examination of licensing procedures should determine whether they are responsive to new information that is developed about plant or operator performance. Effort should also be made to determine whether results of examinations can be correlated with operator ability. Requalification training and testing should be similarly examined to ensure that they take account of information that is developed by operation in the plant and to determine that relevant information about other plants is made available to operators and is made part of the training and requalification program. As part of this and of other more extensive studies, continuing attention must be given to the amount of information which an operator can assimilate and use in normal and in emergency situations and to the best method of presenting the information to the operator. The use and limitations of simulators for operator training should receive careful consideration.

Response

TVA agrees that the Nuclear Regulatory Commission should examine operator qualification requirements, training, and licensing procedures. The NRC and the utilities should work together to determine what changes are needed and to establish uniform performance and training requirement.

Educational background standards and the results of general aptitude tests are utilized by TVA in the selection of candidates for our student operator training program. Although the admission standards increase the probability of a student successfully completing the program, they do not guarantee the development of a better operator. This must be accomplished by establishing a rigorous and comprehensive program with many checks along the way. The TVA program meets these objectives. Our student operator training is 26 months in length and is divided into four sections or student levels. To progress from one level to the next, the student must pass comprehensive written and oral examinations.

The content of the student training program is outlined below:

Student I, Step 1

1. Mathematics
2. Chemistry
3. Physics
4. Thermal Hydraulics

Student I, Step 2

1. Print Reading
2. Introduction to Secondary Plant Cycles and Components
    A. Valves
    B. Pumps
    C. Instrumentation
3. Secondary Plant Cycles

Student II, Step 1

1. Physics (Electricity and Magnetism)
2. Plant Normal and Emergency Electrical Systems

Student II, Step 2

1. Principle of Turbogenerator Operations
2. Turbine Construction
3. Turbine Control Systems
4. Generator Constructions
5. Generator Cooling and Excitations Systems

Student III, Step 1

1. Nuclear Physics
2. Reactor Physics and Operations
3. Introduction to Primary Systems

Student III, Step 2 (Conducted at Plant; Four Months)

1. In-depth study of all plant systems (Construction and Operation)

Student IV

1. On-the-job training. Assigned to shift for five months as student assistant operator.

Following completion of the 26-month training program, the student is reassigned as an assistant unit operator for a minimum of 14 months. During this time the individual must complete a hot or cold licensing program and must pass an operator certification examination which is administered on a simulator identical to the plant at which the assistant unit operator is assigned.

While written examinations are an acceptable method for determining the knowledge level of a student, TVA believes that only through oral and performance examinations on a simulator can an individual's operating ability be properly evaluated.

The TVA requalification program consists of three weeks of retraining. Two weeks of the retraining are conducted on the plant simulator at the Power Production Training Center and one week at the plant. During the simulator training, the operations are continuously evaluated and graded

on their performance during normal and emergency operations. The program is tailored for each plant to ensure that all plant modifications, new plant information, and new operating procedures and techniques are considered as well as other specific plant or utility problems including Licensee Event Reports. The results of the Training Center evaluation are documented, and a report is forwarded to the licensee's plant. All unsatisfactory evaluations are reviewed by the plant Training Review Board which consists of the plant superintendent, assistant superintendent, operations supervisor, and assistant operations supervisor. Based on this review, the Training Review Board establishes additional retraining requirements that must be satisfactorily completed before an operator can resume licensed shift activities.

After completion of retraining, an examination approximately 6-8 hours in length is administered, by the plant, to all licensed operators. If an operator receives a score of less than 80 percent in any category or receives an overall score of less than 70 percent, the Training Review Board reviews the individual's training record and assigns additional retraining in all categories in which a deficiency has been demonstrated. Removal from all licensed activities and placement in an accelerated retraining program is mandatory when an overall score of less than 70 percent is obtained. Before an operator resumes licensed activities, he must demonstrate proficiency in the areas in which he was previously judged deficient and must be approved by the Training Review Board after the Board has conducted a review of the operator's performance in the accelerated retraining program.

In addition to the above, each licensed operator periodically receives a shift performance evaluation. This evaluation is based on the operator's performance on shift during transient, startup, shutdown, emergency, and abnormal conditions.

ACRS Statement--Evaluation of Licensee Event Reports

Because of the potentially valuable information contained in Licensee Event Reports (LER's), the Committee recommends that the NRC Staff establish formal procedures for the use of this information in the training of supervisory and maintenance staffs and in the licensing and requalification of operating personnel at commercial nuclear power plants. The information in LER's may also be useful in anticipating safety problems. At the present time, some utilities routinely request that they be provided copies of all LER's applicable to plants of the type they operate or to specific systems and components in a given class of plants similar to their plant. Certain reactor vendors have made similar requests and use the LER's to review and evaluate the performance of their plants. In addition, the NRC operator licensing staff has indicated that they use LER's in reviewing operating experience at commercial facilities.

The large number of LER's that attribute the cause to personnel error would tend to indicate that a formalized program of LER review would be useful in the training, licensing, and requalification of nuclear power plant personnel. The extent to which such a program could be used to anticipate safety problems should also be considered.

Response

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, monthly 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.

ACRS Statement--Operating Procedures

Sa' ty aspects of individual reactors during normal operation and under accident conditions are reviewed in detail by the NRC Staff and discussed with the ACRS. Acceptable limits for normal operations are formalized by technical specifications, submitted by the licensee, and approved by the NRC Staff. Operating procedures for severe transients have received less detailed review by the NRC Staff. It appears that such procedures would benefit from review by an interdisciplinary team which includes personnel expert both in operations and in system behavior. Also, for the longer term, there may be merit in considering the development of more standardized formats for such procedures.

Response

Sequoyah operating instructions undergo an independent review by the Plant Operations Review C mittee. This Committee consists of the plant superintendent, assistant superintendent, maintenance supervisor, operations supervisor, results supervisor, health physics supervisor, and quality assurance supervisor. In addition, TVA will develop and implement internal procedures to ensure that all abnormal and emergency operating procedures are reviewed by the Division of Engineering Design. This process will ensure that abnormal and emergency operating instructions undergo an interdisciplinary review by personnel expert in both operations and system behavior.

ACRS Statement--Reliability of Electric Power Supplies (System Design and Testing)

During the past several years, there have been several operating experiences involving a loss of ac power to important engineered safeguards. The ACRS believes it important that a comprehensive reexamination be made by the NRC and the reactor licensees of the adequacy of design, testing, and maintenance of offsite and onsite ac and dc power supplies. In particular, failure modes and effects analyses should be made, if not already performed, more systematic testing of power system reliability, including abnormal or anomalous system transients, should be considered, and improved quality assurance and status monitoring of power supply systems should be sought.

Response

The Sequoyah Nuclear Plant is supplied with electrical power from two major systems, an offsite power system and an onsite power system. The offsite electrical power is supplied by two physically and electrically independent connections from the Sequoyah 161-kV switchyard to the onsite electrical distribution system. The 161-kV switchyard is the terminus for the second nuclear unit, the 500-kV intertie bank and nine 161-kV transmission lines.

The onsite power system consists of an ac power system and a dc power system. The onsite ac power system is a Class IE system which consists of: (1) the standby ac power system and (2) the 120-V vital ac system. The standby power system is identified as the diesel generators, the 6.9-kV shutdown boards, the 480-V shutdown boards, and all motor control centers supplied by the 480-V shutdown boards for both units.

The 120-V vital system is powered by four inverters per unit each powering a separate channel through a separate instrument power board. The vital 120-V dc control power system is a Class IE system composed of four redundant channels fed from the four vital batteries.

Prior to operation, functional and preoperational tests are conducted on each system. During the preoperational test phase, system performance is demonstrated during both normal and abnormal conditions. The results of these tests are then reviewed by TVA's Division of Engineering Design.

During plant operation, system testing is controlled by plant technical specification surveillance requirements. If any major maintenance is performed, post-maintenance tests and/or surveillance tests are performed to ensure the operability of the system.

TVA is aware of the importance of the reliability of electric power supplies and periodically reviews the adequacy of both the offsite and onsite ac and dc power supplies. Additional monitoring of the systems during system transients is being considered as well as including the power systems on the emergency core cooling status monitoring system.

ACRS Statement--Analysis of Transients

The ACRS recommends that each licensee and holder of a construction permit be asked to make a detailed evaluation of his current capability to withstand station blackout (loss of offsite and onsite ac power) including additional complicating factors that might be reasonably considered. The evaluation should include examination of natural circulation capability. the continuing availability of components needed for long-term cooling, and the potential for improvement in capability to survive extended station blackout.

The ACRS also recommends that each licensee and construction permit holder should examine a wide range of anomalous transients and degraded accident conditions which might lead to water hammer. Methods of controlling or preventing such conditions should be evaluated, as should research to provide a better basis for such evaluations. The Committee expects it would be appropriate to have such studies done generically first, for classes of reactor designs and system types.

Response

The Sequoyah auxiliary feedwater system includes a separate turbine-driven pump and redundant electric-driven feed pumps to provide motive power diversity for the system. The valves and controls and necessary support systems of the turbine-driven pump are powered by a 1E dc power source. In addition to this major plant feature, Westinghouse has done some preliminary work on this issue. The time that Sequoyah will be able to survive total blackout is largely dependent upon the loss rate of primary coolant through the reactor coolant pump seals. Further study will be necessary to develop a complete picture of plant response to this event. To be most efficient, this work should be done on a generic basis.

TVA agrees that systematic studies of operating and accident situations that might lead to water hammer could be valuable in improving the reliability and safety of power plants. TVA is willing to participate in generic studies of this phenomenon.

ACRS Statement--Emergency Planning

An effort should be undertaken to plan and define the role NRC will play in emergencies and what their contribution and interaction will be with the licensee and other emergency plan participants including other government agencies, industry representatives, and national laboratories. Such planning should consider:

1. Assurance that formal documentation of plans, procedures, and organization are in place for action in an emergency.

2. Designation of a technical advisory team with names and alternates for the anticipated needs of an emergency situation.

3. Compilation of an inventory of equipment and materials which may be needed for unusual conditions including its description, location, availability, and the organization which controls its release.

The Committee recommends that each licensee be asked to review and revise within about three months:

1. His bases for obtaining offsite advice and assistance in emergencies, from within and outside the company.

2. Current bases for notifying and providing information to authorities offsite in case of emergency.

This review and evaluation should be in terms of accidents having a broad range of consequences. The results of this review should be reported to the NRC.

Response

TVA has reviewed the bases for obtaining offsite advice and assistance in emergencies. TVA's Radiological Emergency Plan (REP) identifies the bases and mechanisms for requesting assistance for emergencies requiring offsite response.

TVA also has reviewed its current bases for notifying and providing information to offsite authorities. The TVA REP provides criteria and a flowchart for notifying and providing information to the lead State agency, NRC, and DOE for all potential or accidental radiation releases.

ACRS Statement--Decontamination and Recovery

The Committee wishes to call attention to the importance of a program designed to learn directly about the behavior, failure modes, survivability, and other aspects of component and system behavior at TMI-2 as part of the long-term recovery process. This program should also examine the lessons learned at TMI-2 to determine if design changes are necessary to facilitate the decontamination and recovery of major nuclear power plant systems.

Response

The long-term recovery effort at TMI-2 has been and will continue to be a valuable classroom for the industry. This experience should continue to yield insights and data that will be useful in the evaluation and design of plant systems for accident recovery.

ACRS Statement--Safety Review Procedures

The TMI-2 accident has imposed large new pressures on the availability of manpower resources within the NRC Staff. If progress is to be expedited on the new questions which have arisen and on existing unresolved safety issues, the ACRS believes that new mechanisms should be sought and implemented. For those safety concerns where such a mechanism is appropriate the Committee recommends that the Commission should request licensees to perform suitable studies on a timely basis, including an evaluation of the pros and cons, and proposals for possible implementation of safety improvements. The NRC Staff should concurrently establish its own capability to evaluate such studies by arranging for support by its consultants and contractors. In this fashion, the Committee anticipates that the information on which judgments will be based can be developed much more expeditiously, and an earlier resolution of many safety concerns may be achieved.

Response

Not applicable.

ACRS Statement--Capability of the NRC Staff

The Committee recommends that the capability of the NRC Staff to deal with basic and engineering problems in what may be termed broadly as reactor and fuel cycle chemistry be augmented expeditiously. This should include establishment of expertise within the NRC, with assistance arranged from consultants and contractors, in such important technical areas as the behavior of PWR and BWR coolants and other materials under radiation conditions; generation, handling, and disposal of radiolytic or other hydrogen at nuclear facilities; performance of various chemical additives in containment sprays; processing and disposal techniques for low- and high-level radioactive wastes; chemical operations in other parts of the nuclear fuel cycle; and in the chemical treatment operations involved in recovery, decontamination, or decommissioning of nuclear facilities. The Committee wishes to emphasize the importance of providing this expertise in both the research and licensing management elements of the NRC.

Response

Not applicable.

ACRS Statement--Single Failure Criterion

The NRC should begin a study to determine if use of the single failure criterion establishes an appropriate level of reliability for reactor safety systems. Operating experience suggests that multiple failures and common mode failures are encountered with sufficient frequency that they need more specific consideration. This study should be accompained by concurrent consideration of how the licensing process can be modified to take account of a new set of criteria as appropriate.

Response

Not applicable.

ACRS Statement--Safety Research

The ACRS believes that, as a result of the TMI-2 accident various safety research areas will warrant initiation or much greater emphasis, as appropriate. The Committee suggests that consideration be given to an augmentation of the NRC safety research budget for FY 80.

Also, the Committee believes that a larger part of the safety research program should be oriented toward exploratory research as contrasted to confirmatory research, with some degree of freedom from immediate licensing requirements. The ACRS plans to have a Subcommittee meeting on this subject with representatives of the NRC Office of Nuclear Regulatory Research in the near future.

The Committee is continuing to review these matters and will report further as additional recommendations are developed.

Response

Not applicable.

Additional Comments by Messrs. H. Lewis, D. Moeller, D. Okrent, and J. Ray

The potential for a reduction in risk to the public in the case of a serious reactor accident by the implementation of a means for controlled, filtered venting of a containment which could retain particulates and the bulk of the iodine has been recognized for more than a decade. The concept was recommended for study more recently in the American Physical Society Report on light-water reactor safety and in the Ford Foundation-Mitre Report, "Nuclear Power - Issues and Choices." It is a high priority item in the NRC plan submitted to Congress for Research to Improve the Safety of Light-Water Nuclear Power Plants (NUREG-0438). The study performed for the State of California on underground siting concluded that filtered, vented containment was a favored option to explore in connection with possible means to mitigate the consequences of serious reactor accidents. However, little progress has been made on the development of sufficiently detailed design information on which to evaluate the efficacy and other factors relevant to a decision on possible implementation of such consequence ameliorating systems.

The TMI-2 accident suggests that the probability of a serious accident in which a filtered, vented containment could be useful is larger than many had anticipated.

We recommend that the Commission request each power reactor licensee and construction permit holder to perform design studies of a system which adds the option of filtered venting or purging of containment in the event of a serious accident. The system should be capable of withstanding a steam and hydrogen environment and of removing and retaining for as long a time as necessary radioactive particulates and the great bulk of the iodine for accidents involving degraded situations up to and including core melt. Such studies could be done generically for several reactor-containment types, and should evaluate the practicality, pros and cons, the costs, and the potential for risk reduction. A period of about 12 months for a report to the NRC by licensees and construction permit holders appears to represent a possible schedule.

Response

TVA has been involved in a number of studies of advanced containment concepts over the years. Although the vented and filtered concept has merit, it is not clear that this particular approach will always be the most efficient if additional containment capability is determined to be necessary. TVA is willing to participate in industry studies to evaluate the need for enhanced containment capability and to consider design proposals.