Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

October 2, 2000

10 CFR 50.54 (f)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

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

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) (TAC NOS. M83674 and M83675)

Reference: NRC Letter to TVA dated August 2, 2000, "Sequoyah Nuclear Plant, Units 1 and 2 - Request For Additional Information On Individual Plant Examination Of External Events (TAC No. M83674 and M83675)"

The purpose of this letter is to provide our response to NRC's request for additional information relative to the above reference. Specifically, the enclosure provides our response to the "Fire Events" questions only. Our response for the questions pertaining to "Seismic Events" of the above reference will be enclosed under a separate letter.

A011

If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 2nd day of October

Peggy M. Billingsley
Notary Public

My Commission Expires Oct. 9, 2002

Enclosure
cc: See page 3

cc (Enclosure):

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2- RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) (TAC NOS. M83674 and M83675)

Fire Events

1. In Step 3 of Phase II (please refer to Section 5 of the SQN fire IPEEE summary report), the submittal has introduced a "severity factor" to compute the conditional probability of damage to safe shutdown cables and equipment in a compartment. This "severity factor" is then multiplied by the failure probabilities of automatic and manual suppression. As discussed in Reference [1], severity factors should not be multiplied by non-suppression probabilities, since the two probabilities are based on a common pool of event data. That is, the potential for a large fire is dependent upon failure of fire suppression; therefore, the methodology employed in IPEEE fire analysis effectively results in double counting suppression efforts.

   For each case that the automatic fire suppression was credited in conjunction with the fire severity factors, please explain why such credit does not constitute double counting for suppression. Please note that a re-analysis should be provided, if the multiplication of the severity factors and the non-suppression probabilities cannot be adequately justified.

   RESPONSE:

   For areas outside the main control room (MCR), and as described in Section 5 of Report SQN-IPEEE-005, (section of the revised fire IPEEE submitted on September 1, 1999), the fire severity factors were selected based only on fires where neither hose streams nor automatic suppression were required to suppress the fire. This was done to avoid generating an overall damage factor that could potentially include fires suppressed by the fire brigade or installed systems.

   In other words, a fire that was categorized as *minor* was of such a trivial nature that it was either self-extinguishing or was suppressed by the personnel in the area without hose streams. This methodology avoids the potential for *double counting suppression efforts*. This is shown in the event trees for the various areas, where a minor fire, such as one that was suppressed before the fire brigade responded, would not have the potential to impact a significant amount of area equipment. Additionally, since all industry fires were eventually suppressed, no failures to suppress major fires by the fire brigade have been recorded.

   In its discussion of severity factors, Appendix M of EPRI's December 1995 Final Report, "Fire Probalistic Risk

Assessment (PRA) Implementation Guide," makes clear that the control room is a special case. For other plant areas where suppression is modeled, the suppression system reliability is taken from a different data source than fire severity.

Using EPRI's "Guidance for Developing of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE)" (May 1999) the fire severity factor of 0.15 was removed subsequently increasing the area core damage frequency(CDF) such that recovery of the main control room is required. The above fire PRA implementation guide provides guidance to the effect that, while control room abandonment must be assumed (primarily for habitability and visibility issues as found by Sandia Lab test results) for any fires that are not suppressed within 15 minutes, control room recovery at 60 minutes can be incorporated, if necessary.

Based on review of the SQN's "Control Room Inaccessibility" Abnormal Operating Procedure, Rev 2, July 10, 2000, Section 2.3, which provides guidance on recovery of the control room, a screening value of 0.10 is used. Due to the more liberal timing available and knowing what plant control functions (i.e., reactor coolant system (RCS) makeup, auxiliary feedwater (AFW) flow, etc.) need to be recovered, this value is judged to adequately bound the actual value. That is, the recovery of the main control room (MCR) is judged to be no more rigorous an action than control room abandonment, which is evaluated with a failure rate of 0.074 (value taken from NUREG evaluation for reference plants). Therefore, a human error probabilities (HEP) of 0.1 is used for this action which is slightly greater HEP than the HEP of 0.074 used for the control room abandonment action.

Therefore, the frequency for the primary scenario of concern becomes:

$$1.93E-2 \times 0.0034 \times 0.074 \times 0.10 = 4.86E-7$$

Where,

1.93E-02	is the total control room fire ignition frequency for the Unit 1 and 2 control areas.
0.0034	is the fire suppression value given in the Fire PRA Implementation Guide.

0.074	is the failure rate for control room evacuation (i.e. loss of at least one key function during shift of plant control from the main control room to the alternate shutdown panel.
0.10	is the failure rate for recovery of key control room functions after 60 minutes.

For completeness, the core damage frequency for suppressed fires can be calculated as:

1.93E-02 x 0.9966 x 3.30E-06 = 6.35E-08

Where,

1.93E-02	is the total control room fire ignition frequency for the Unit 1 and 2 control areas.
0.9966	is the successful fire suppression term (i.e., 1.0 - 0.0034).
3.30E-06	is the conditional core damage frequency, given successful suppression (similar to a total loss of main feedwater - see Case 1 in the initial main control room evaluation).

In event tree layout, this would be

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             Fire Suppressed
1.93E-02 ----------------------------------------  6.35E-08
          |  Fire not suppressed (CR Evacuation)
          ----------------------------------------  4.86E-07
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In summary, for areas outside the MCR there was no double counting suppression efforts. In the MCR, the screening evaluation has been modified, as discussed above, to account for control room recovery which results in the CDF for the MCR being below the screening criteria.

2. Human error probabilities have not been discussed in the fire IPEEE submittal. However, based on the revised IPEEE submittal it is not clear if the licensee has properly modified the individual plant examination (IPE) human error probabilities (HEPs) to reflect the special conditions imposed by the occurrence of a fire. It is important that the HEPs properly reflect the potential effects of fire (e.g., smoke, heat, loss of lighting, etc.), even if these effects do not directly cause damage to equipment in the scenarios being analyzed. If the influence of fire environment on quantifying the various HEPs is not properly considered, there is a potential for assigning optimistic HEP values. Note that HEPs which are conservative with respect to an internal events analysis could be non-conservative with respect to a fire risk analysis.

Please provide:

(a) A list of compartments which were screened-out from further analysis in Steps 2 and 3 of Phase II, whose quantification involved one or more HEPs;

(b) The description and numerical values of HEPs for each of these compartments; and

(c) The procedure used to quantify the impact of postulated fires on the various HEPs.

Please note that a reanalysis should be provided if the influences of fire (i.e., smoke, loss of lighting, poor communication, etc.) were not taken into account, where relevant, in establishing the performance shaping factors (PSFs).

RESPONSE:

The list of rooms screened from further evaluation (in steps 2 and 3 of Phase II) whose quantification of core damage probability (CDP) incorporated any operator are found in Tables-4.2, -5.1 and -5.2 of the SQN IPEEE (Fire) Summary (section of the revised fire IPEEE submitted on September 1, 1999).

In step 1 of Phase II (the initial screening level, described in Section 3.2 of the SQN-IPEEE-003 report, submitted with the revised fire IPEEE on September 1, 1999) the base-risk model was modified to fail operator (recovery) actions for plant functions that were not specifically identified as being available following a fire. These top events include all ventilation recovery actions (top events

VTxxR), secondary plant cooldown with steam dumps (top event CD) and main feedwater recovery (top event OF). In general, the fires of concern would potentially impact the equipment being manually operated, such that the scenarios of concern for this comment would be those that a fire would impede or prevent the operator from reaching the unaffected compartment from which to perform the action (i.e., hardware failure at the top event level would prevent operator success). Operator actions in the base model, outside the MCR, which could potentially be affected by a fire are: recovery of the turbine-driven (TD) auxiliary feedwater pump and alignment of the spare battery charger. These operator actions are discussed below:

Recovery of Turbine Driven AFW Pump 1A-S

One human action that was credited at the initial screening level involves the recovery of the TD AFW pump. This was applied to areas where the TD AFW pump hardware was not impacted by the fire (i.e., the fire-hazard analysis confirmed that there were no associated components or cables in the area). Failure of this recovery was incorporated as stated in Section 3.3 (bottom of page 18 of SQN-IPEEE-003). For this recovery action, the failure rate is 0.808 (split fraction TPR1), or success of less than 1 in 5 attempts. In general, the operator is prevented from recovering due to the failure mode of the TD AFW pump. This extremely high failure rate was judged to adequately bound the conditions where the operator would have to transit through areas of potential confusion while attempting to perform the action areas.

For reference, (see event tree rule for interim variable TPFIRE on page 36 of SQN-IPEEE-003), top event TPR is set to guaranteed failure for auxiliary building fire areas 001, 012, 070, and 085. As shown in Table 4 of SQN-IPEEE-003, two of these areas were screened at the initial level of evaluation (FAA-001 and FAA-012). For the remaining 68 areas that were screened, TPR was allowed. Review of the screened areas confirm that the operator would not be required to transit through these areas to reach the turbine driven AFW pump

Alignment to Alternate Battery Charger

The detailed evaluation of Area 734.0-A.01 is described in Section 3.4.3 of report SQN-IPEEE-005. This section relaxes the conservative assumption of outright failure of 125-VDC buses II and III, since the fire only has the potential to impact the battery chargers for these buses. Therefore,

recovery was incorporated through use of split fractions DB3 and DC3, instead of guaranteed failure, for these top events. This section of the report then goes on to state:

> *"The operator action to shift the battery supply to the alternate charger is directed by the site Fire Interaction Manual (0-GO-8, Reference 16). This long term (required to be performed within a several hour time window, prior to battery discharge) is performed outside the affected area. Therefore, the currently modeled operator action is judged to adequately bound the conditions required following a potential fire."*

In summary, these operator actions were considered credible and the numeric value of their HEPs were 0.808 for recovery of the turbine driven pump and 0.018 for the alignment of the spare battery charger, respectively. In addition, the above discussions provide the basis for the effect of fires on the HEPs as requested in Part (c) of Question No. 2.

The effect of a fire on operator actions, accomplished from the MCR are discussed below:

Control Room Actions

Discussion with the NRC reviewers during a telephone conference held on June 15, 2000, revealed the concern that control room actions would also potentially be impacted by the fire. In general, these actions, particularly at the initial screening level, addressed operator recovery from hardware failure, such as initiating a manual reactor trip, following failure of the actuation logic. Review of the descriptions for these actions confirm that the conditions for which the actions are evaluated adequately bound the case where a fire in the plant could potentially divert a portion of the control room team attention.

It should be noted that, at both levels of screening, the fire evaluation assumed that plant trip would occur in response to all plant fires. Step 9 of the SQN's, "Plant Fire," Abnormal Operating Procedure, Rev. 8, August 16, 2000, has the operator verify that safe reactor operation can be maintained (i.e., plant shutdown may be required due to loss of component operability or if continued plant operation would hinder fire fighting efforts). If not, the operator is directed to evaluate the need for reactor trip or emergency shutdown and perform plant shutdown using appropriate procedures, which are then listed.

At the detailed screening level, the most significant operator action involves initiation of bleed and feed cooling. The incorporation of this action is described in detail in response to Question No. 3. For the initial screening evaluation described in SQN-IPEEE-003, this function was set to guaranteed failure at the top event level.

In summary, the effect of a fire on the HEPs for operator actions performed in the MCR (i.e., manual reactor trip, etc.) and why these remaining actions would not be affected by a fire is discussed above. These remaining operator actions, used in the quantification of CDP for Phase II steps 2 and 3, are taken directly from the SQN PSA and are listed below along with their HEPs:

Manual Reactor Trip - 0.005
Initiate Feed and Bleed Cooling - 0.0145

3. Related to the request for additional information (RAI) number 2 above, the human error probabilities used in computing the conditional core damage probabilities (CCDPs) for bleed and feed appear to be optimistic. The core damage frequency (CDF) of several fire scenarios (e.g., Essential Raw Cooling Water Intake Pumping Station rooms 704.0-E01 and 720.0-E01, Auxiliary Building rooms 714.0-A03, 714.0-A05, 714.0-A05', 714.0-A09, 734.0-A13, 734.0-A28, 749.0-A01, 749.0-A04, 759.0-A01, etc.) were reduced by several orders of magnitude in the final screening step. Since bleed and feed mode of core cooling requires manual operator actions, the estimated small CCDP implies the use of very small human error probabilities (HEPs) or multiplication of several HEPs without taking into consideration the possibility of dependencies among various human actions.

For scenarios where feed and bleed was credited, please provide the basis for obtaining small CCDPs, including the basis for those human error probabilities that have the largest impact on the calculated CCDPs. If HEPs are multiplied, please provide the basis for assuming independence among the various human actions. Please note that a re-analysis should be provided, if the technical basis for various HEPs and their independence cannot be adequately justified.

RESPONSE

A large decrease in CDF for several plant areas by incorporating into the analysis optimistic HEPs for operator

action to initiate feed and bleed cooling is not the case. The failure rate for bleed and feed cooling is given in the SQN IPE Rev 1, September 1995, as 2.66E-2 (split fraction TPR1). Of this failure rate, 1.45E-2 represents operator failure where the remainder is due to component hardware failures. This operator action is proceduralized and the operators are regularly trained in scenarios requiring this action in the control room simulator. This action takes place following a loss of secondary heat removal (i.e., failure of turbine and motor driven AFW). In addition, HEPs were not multiplied to obtain this scenario frequency, such that this represents the failure rate for establishing bleed and feed cooling as a single, independent human action.

The human action evaluation methodology used for Sequoyah in the original IPE submittal was consistent with other IPEs methodology findings reviewed by NRC. This methodology is based on the Success Likelihood Index Methodology (i.e., SLIM-MAUD) where various performance shaping factors are evaluated for each action and is described in detail in Appendix B of the IPE report.

The reductions in CDF in the Phase II, step 3 evaluations were primarily obtained by further analyses which demonstrated that equipment which was assumed failed in the Phase II, step 2 evaluations would be available for the plant areas in question. Each of these evaluations is described in the related text in the submittal. For example, areas 714.0-A03, -A05, -A05′ and -A09 are described in Section 3.3.2 of SQN-IPEEE-005. Particularly, the initial evaluation of areas 714.0-A05 and A05′ assumed outright failure of 6.9-kV shutdown boards 1A-A and 2A-A, whereas review of the actual cable routing confirmed that only the backup supply to the boards from the diesel generators could be impacted by a fire in this area. When taken with the availability of bleed and feed cooling, this reduced the conditional core damage frequency significantly (from 6.6E-2 to 8.0E-5 for area 714.0-A05).

For each area of concern, a summary of the equipment availability changes made in the Phase II, step 3 evaluations is provided below. For further details, these summaries reference the applicable section of SQN-IPEEE-0005.

<u>Essential Raw Cooling Water (ERCW) Intake 704.0-E01</u> previously assumed guaranteed failure while performing maintenance on a single pump, effectively reducing one train of ERCW to a 2 of 2

system. This was modified as described in the text to model 2 of 3 available pumps when performing maintenance. See Item 1 under Section 2.1 of SQN-IPEEE-005.

ERCW Intake 720.0-E02 previously assumed that all four train B pumps failed, whereas spatial separation with no significant intervening combustibles prevents total failure. See Item 2 under Section 2.1 of SQN-IPEEE-005.

Auxiliary Building 714.0-A03 previously assumed the loss of ERCW cooling to all diesel generators and failure of component cooling pumps 1A-A and 1B-B. These cables are in a protected pull box in the corner of the area. Also, zone of influence calculations show that there are no credible fires that could impact component cooling water system cables in the area. See Section 3.3.2 of SQN-IPEEE-005.

Auxiliary Building 714.0-A05 previously assumed guaranteed failure of 6.9-kV shutdown boards 1A-A and 2A-A. This was relaxed to only failure of the diesel generator supply cables to each of the shutdown boards. See Section 3.3.2 of SQN-IPEEE-005.

Auxiliary Building 714.0-A05' previously assumed guaranteed failure of 6.9kV shutdown boards 1A-A and 2A-A. This was relaxed to only failure of the diesel generator supply cables to each of the shutdown boards. See Section 3.3.2 of SQN-IPEEE-005.

Auxiliary Building 714.0-A09 previously assumed guaranteed failure of 6.9-kV shutdown boards 1B-B and 2B-B. This was relaxed to only failure of the diesel generator supply cables to each of the shutdown boards. See Section 3.3.2 of SQN-IPEEE-005.

Auxiliary Building 734.0-A13 previously assumed guaranteed failure of two 6.9-kV shutdown boards. This was relaxed to fail the diesel generator supplies to the boards only. See Section 3.4.2 of SQN-IPEEE-005.

Auxiliary Building 734.0-A28 previously modeled a total loss of site offsite power. This was relaxed to a loss of offsite feed to bus 1B-B only. See Section 3.4.2 of SQN-IPEEE-005.

Auxiliary Building 749.0-A01 previously assumed guaranteed failure of 6.9-kV shutdown boards 1A-A and 2A-A. This was relaxed to only failure of the diesel generator supply cables to each of the shutdown boards. See Section 3.5.3 of SQN-IPEEE-005.

Auxiliary Building 749.0-A04 Review of the IPEEE Fire Submittal for this response revealed that feed and bleed cooling, using two pressurizer power operated relief valves (PORVs) and one high head pump would not be possible for a fire in this battery board room because loss of the battery board disables the associated PORV.

This room is reevaluated here using a better estimate success criteria for feed and bleed cooling which only requires one PORV. This success criteria is consistent with information provided as part of the event tree development for the NRC's Significance Determination Process. Using the one PORV success criteria and failing the PORV associated with the battery board in this room, results in a conditional core damage frequency of 1.24E-4. Total core damage frequency for this area is then 1.21E-3 x 1.24E-4 = 1.50E-7. While this value is higher than reported in SQN-IPEEE-005, it remains well below the screening cutoff of 1E-6.

Auxiliary Building 759.0-A01 previously assumed guaranteed failure of 6.9-kV shutdown boards 1A-A, 1B-B and 2A-A. This was relaxed to only failure of the diesel generator supply cables to each of the shutdown boards. See Section 3.6 of SQN-IPEEE-005.

Other areas are similarly discussed in SQN-IPEEE-005.

4. Based on the information provided in the IPEEE submittal, it can be inferred that smoke from several compartments can enter the Main Control Room. For example, on page 89 of Reference [2] it is stated that room 749.0-A15 includes the

Main Control Room air handlers. It may be inferred that a fire in those compartments may lead to smoke in the Main Control Room. The submittal does not provide an explicit discussion of the impact of smoke due to sources in other compartments on the Main Control Room.

Please provide a description of the analysis conducted to justify the IPEEE exclusion of the potential for smoke ingress into the Main Control Room. Please note that a re-analysis should be provided, if such an analysis was not conducted, which includes prevention of smoke ingress into the main control room or the abandonment of the main control room, as appropriate.

RESPONSE:

Smoke ingress into the MCR was not considered in the IPEEE FIRE analyses. This decision was based on discussions, with the site fire protection engineer, which confirmed that the MCR is protected from intake of fire combustion products from other plant areas by fire dampers that are actuated by ionization detectors.

In addition, Step 6 of SQN's, "Plant Fire," Abnormal Operating Procedure, Rev 8, August 16, 2000, has the operator confirm that smoke is not entering the MCR. If so, the operator is directed to isolate control room ventilation and place CREVS in service. Once started, control room emergency ventilation system CREVS is adequate to maintain an acceptable environment in the control room.

Also the site prefire plans have a section devoted to Ventilation Methodology. For the area of interest, prefire plan AUX-0-749-01, directs the operator to consider shutting down air handling units to prevent smoke spreading. The operator is also directed to see procedure 0-SO-30-11, which provides specific direction on operation of the air handling units. Restoration of control building ventilation is then addressed in procedure 0-SO-30-1.

Only one of the MCR air handling units (B-B) can be impacted by a fire in the area noted in question no. 4. Unit A-A (see evaluation of area 734.0-A07 on page 73 of SQN-IPEEE-003) remains unaffected.

These plant features, in addition to the potential for manual operator action to prevent smoke intrusion into the MCR, adequately justify the IPEEE exclusion of the potential for smoke ingress into the MCR.

In summary, the MCR is by design and operating procedures protected from being filled with smoke from other areas of the control building.

5. Based on the information provided in the IPEEE submittal, it can be inferred that in addition to the Main Control Room and the cable spreading room, there are several other areas in the plant where components and cables from both units are co-located (shared compartments). A list of the shared compartments between the two units is not provided in the IPEEE submittal.

   Please provide:

   (a) a list of shared compartments, and

   (b) a discussion of the results of an analysis of the contributions to the estimated fire core damage frequency (for both units) from core damage scenarios where fires are initiated in compartments with components for both units.

RESPONSE:

While there are several areas at SQN where a potential fire could result in a trip of units 1 and 2, each unit has its own resources, such that they would not be competing for a single system, train, or component. For those components that support shutdown of both units (i.e., DC trains I, II, III and IV, etc.), the system is designed with the capability to support both units simultaneously.

The following shared areas exist between SQN Units 1 and 2, in addition to the MCR, cable spreading room, relay room (732.0-C13), and auxiliary control room:

Auxiliary Building common areas - extremely large (25,000 to 30,000 square foot) areas with unit 1 and unit 2 equipment, such as motor driven AFW pumps with large amount of separation between components for each unit.

ERCW intake - a fire in this area that disables an entire train of ERCW (i.e., 2 unit 1 and 2 unit 2 pumps for each train) would result in a plant trip of both units due to loss of RCP cooling. Loss of train A of ERCW would result in the loss of component cooling train A and thermal barrier

cooling to both units (note that the Train A ERCW pumps are segregated into a Unit 1 bay and a Unit 2 bay, precluding a single fire from failing all four pumps).

In general, a fire will have a more significant impact on one unit or the other (i.e., shutdown board room 1A, as opposed to 2A). For the Fire IPEEE Analysis, where an area contained Unit 2 components that are not addressed in the Unit 1 model, the impact was modeled as affecting Unit 1. In this way, potential fires were modeled in such a way as to generate a conservative (i.e., higher CDF) number.

The fire evaluation of the control room does address the potential for evacuation due to fire in the unit 1 control area, for which unit 1 controls would potentially be damaged, as well as evacuation due to a fire in the unit 2 control area due to atmospheric contamination (i.e., smoke, combustion products). This evaluation provides a bounding value for fire in each of the control areas. The fire-related CDF for unit 2 would be similar to that for unit 1 (i.e., the fire-related CDF would be additive, rather than multiplicative).

In summary, the screening evaluations always used:

- the higher of either the unit-1 or unit-2 fire ignition frequency for unit specific plant areas and

- the CDF from the unit which was most adversely affected by a fire in a given plant area.

Therefore, the screening evaluation is bounding and applicable to both units.

References

1. "Guidance for Development of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE)," Prepared by Data Systems and Solutions, LLC for Electrical Power Research Institute, May 1999.

2. Sequoyah Nuclear Plant, "IPEEE Fire-Induced Vulnerability Evaluation (FIVE) - Determination of Fire Scenario Safe Shutdown Path Unavailability," Report SQN-IPEEE-003, May 1999.

3. Sequoyah Nuclear Plant, "Individual Plant Examination for External Events - Fires -Quantitative Screening - Phase 2 (Detailed Screening)," Report SQN-IPEEE-005, RIMS B38 990611 805.

4. "Fire PRA Implementation Guide," EPRI TR-105928, Prepared by Science Applications International Corporation for the Electrical Power Research Institute, Final Report, December 1995.

5. Sequoyah Nuclear Plant, "Control Room Inaccessibility," Abnormal Operating Procedure AOP-C.04, Revision 1, September 9, 1998.

6. Sequoyah Nuclear Plant, "Plant Fires," Abnormal Operating Procedure AOP-N.01, Revision 7, January 20, 2000.