TENNESSEE VALLEY AUTHORITY
KNOXVILLE, TENNESSEE 37902

400 Chestnut Street Tower II

July 31, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Schwencer:

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

Enclosed is information requested by J. Buzy of your staff with regard to NUREG-0694 Items I.A.2.1, I.A.2.3, and I.A.3.1. These items are related to revisions to the Sequoyah Nuclear Plant Operator Training Program as a result of the Three Mile Island accident.

If you have any questions, please get in touch with D. L. Lambert at FTS 857-2581.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure

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ENCLOSURE

SEQUOYAH NUCLEAR PLANT
RESPONSE TO NUREG-0694
ITEMS I.A.2.1, I.A.2.3, AND I.A.3.1

I.A.2.1. (Paragraph 4)

Revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

Response:

TVA's DPM No. N78A13 (June 1980 draft) lists the following requirements for the cold license, hot license, and the requalification training program.

> Related technical training for trainees in the License Programs at the RO level cover the following general subjects:
>
> 1. Principles of reactor operation.
> 2. Design features of the nuclear power plant involved.
> 3. General operating characteristics of the nuclear power plant involved.
> 4. Instrumentation and control systems.
> 5. Safety and emergency systems.
> 6. Standard and emergency operating procedures.
> 7. Radiation control and safety provisions.
> 8. Principles of heat transfer and fluid mechanics.
>
> In addition to the above subjects, related technical training for the trainees at the SRO level cover the following general subjects:
>
> 1. Reactor theory.
> 2. Handling and disposal of, and hazards associated with, radioactive materials.
> 3. Specific operating characteristics of the nuclear power plant involved.
> 4. Fuel handling and core parameters.
> 5. Administrative procedures, conditions, and limitations.
> 6. Theory of fluids and thermodynamics.

> Emphasis shall be placed on reactor and plant transients and training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

A special 14-hour program "Training For Mitigating Core Damage" was developed by TVA and submitted to the NRC in July 1980. All Sequoyah licensed operators will have completed this course by August 1, 1980.

I.A.2.3 (Paragraph 3)

Instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

Response:

TVA's DPM No. N78A13 (June 1980 draft) lists the following instructor qualifications:

> All instructors shall be enrolled in appropriate requalification programs to ensure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations.

I.A.3.1. (Paragraph 2)

Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

Response:

See I.A.2.1 above.

I.A.3.1 (Paragraph 4)

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

Response:

TVA's DPM No. N78A13 (June 1980 draft) states the following requirements.

> A score of 70 percent on each category of the annual written examination is considered passing; however,

an average score of 80 percent or above for all categories of the examination must be maintained by the licensee. If a licensee scores below 70 percent on any category or averages below 80 percent overall, this individual is removed from license activities and placed in accelerated training.

I.A.3.1 (Paragraph 6)

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

Response:

TVA's DPM No. N78A13 (June 1980 draft) states the following requirements.

Control Manipulations

The following control manipulations and plant evolutions where applicable to the plant design are acceptable for meeting the reactivity control manipulations required by Appendix A, Paragraph 3.a., of 10 CFR Part 55. The starred items shall be performed on an annual basis; all other items shall be performed on a two-year cycle. However, the requalification programs shall contain a commitment that each individual shall perform or participate in a combination of reactivity control manipulations based on the availability of plant equipment and systems. Those control manipulations which are not performed at the plant may be performed on a simulator. The use of the technical specifications should be maximized during the simulator control manipulations. Personnel with senior licenses are credited with these activities if they direct or evaluate control manipulations as they are performed.

*(1) Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.

(2) Plant shutdown.

*(3) Manual control of steam generators and/or feedwater during startup and shutdown.

(4) Boration and/or dilution during power operation.

*(5) Any significant (> 10%) power changes in manual rod control or recirculation flow.

(6) Any reactor power change of 10% or greater where load change is performed with load limit control or speed control is on manual.

*(7) Loss of coolant including:
1. Significant PWR steam generator leaks
2. Inside and outside primary containment
3. Large and small, including leak-rate determination
4. Saturated reactor coolant response (PWR)

(8) Loss of instrument air (if simulated plant specific).

(9) Loss of electrical power (and/or degraded power sources).

*(10) Loss of core coolant flow/natural circulation.

(11) Loss of condenser vacuum.

(12) Loss of service water if required for safety.

(13) Loss of shutdown cooling.

(14) Loss of component cooling system or cooling to an individual component.

(15) Loss of normal feedwater or normal feedwater system failure.

*(16) Loss of all feedwater (normal and emergency).

(17) Loss of protective system channel.

(18) Mispositioned control rod or rods (or rod drops).

(19) Inability to drive control rods.

(20) Conditions requiring use of emergency boration or standby liquid control system.

(21) Fuel cladding failure or high activity in reactor coolant or offgas.

(22) Turbine or generator trip.

(23) Malfunction of automatic control system(s) which affect reactivity.

(24) Malfunction of reactor coolant pressure/volume control system.

(25) Reactor trip.

(26) Main steam line break (inside or outside containment).

(27) Nuclear instrumentation failure(s).