

- (4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
 - (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.
- C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 311 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

- a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;
- b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;
- c. Performance of any test at power level different from there described; and

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"
6. WCAP-10266-P-A, "The 1981 Revision of Westinghouse Evaluation Model Using BASH CODE"
7. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"
8. BAW-10186-A, "Extended Burnup Evaluation"
9. EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, "RCS Pressure and Temperature (P/T) Limits"

Specification 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System"

6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."
2. Westinghouse Topical Report WCAP-15321, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."
3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

STEAM GENERATOR (SG) TUBE INSPECTION REPORT

6.9.1.16.1 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.k, "Steam Generator (SG) Program." The report shall include: